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MATERIALS INTEGRITY SECTION
A REVIEW OF 1989 AND PROPOSALS FOR 1990

Report No. 90-73-K

R.G. Fleck
Supervising Engineer
Materials Integrity Section
Metallurgical Research Department

ABSTRACT

This report describes the work programs performed in 1989 by the Materials Integrity Section of the Metallurgical Research Department. The direction of the 1990 work programs is outlined.

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84-875-0200	823.1	May 3, 1990	90-73-K

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EXECUTIVE SUMMARY

MATERIALS INTEGRITY SECTION:

A REVIEW OF 1989 AND PROPOSALS FOR 1990

R.G. Fleck

The programs in the Materials Integrity Section are mainly concerned with providing data to predict the service life of CANDU fuel channels and in particular the pressure tubes. Funding for the studies principally (80%) comes from the CANDU Owners Group.

Decisions on when LSFCR should be performed on Pickering and Bruce reactors have used input from the hydride blister program. From this program a probabilistic code was developed to predict when blisters would form in reactors where pressure tube/calandria tube contact occurred. This code is used to guide decisions on when LSFCR should be performed to avoid the possibility of a Pickering 2 G16 type failure.

The resistivity technique for the nondestructive measurement of hydrogen in pressure tubes was successfully used with irradiated material. Development of a modified instrument for use on samples from the LSFCR of Pickering 3 is underway. The use of differential scanning calorimetry for hydrogen measurements has been demonstrated as a potentially useful technique. This technique is being assessed for the measurement of hydrogen concentration in scrape samples.

Irradiation creep and growth tests were started in the Osiris reactor at Saclay, France. This is a high flux reactor that will provide end-of-life fluences in specimens by the end of the 1990's. From the first growth measurements the data appears comparable to that from the Dido reactor. Dido has been used for about the last 10 years for irradiation growth tests on pressure tube materials but does not have as high a fast neutron flux as Osiris.

Two members of the Section were part of a Canadian delegation to the USSR. Discussions were held on the Canadian and USSR experience with Zr-2.5 wt% Nb pressure tubes. The USSR developed this alloy and are the only other country in the world that has an extensive research program on Zr-2.5 wt% Nb. Further discussions with the USSR are planned.

In support of thermal plants the corrosion/fatigue facility has been upgraded. This facility has been successful in producing service type cracks in boiler tubes.

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TO: D. Mills
Director of Research

MATERIALS INTEGRITY SECTION
A REVIEW OF 1989 AND PROPOSALS FOR 1990

1.0 INTRODUCTION

Throughout 1989 the programs in the Materials Integrity Section continued to be extensively funded through the CANDU Owners Group (COG). Approximately 80% of the budget for the Section comes from COG. Since the formation of the Section, the main thrust of the work programs is related to pressure tubes and covers the areas of Delayed Hydride Cracking, Irradiation Damage and Deformation and Improved Components. There are also studies on fuel cladding, funded by COG, to establish the mechanical properties of oxidized material as a result of exposure to simulated "off normal" conditions. In support of thermal plants, a facility to evaluate the corrosion/fatigue of boiler tubes has been developed under EPRI funds.

The staff of the Materials Integrity Section were extensively involved with the Fuel Channel Seminar which was held in November. Several papers were presented which will be published in the conference proceedings. Two members of the Section participated in a Canadian/USSR information exchange meeting. The USSR use Zr-2.5 wt% Nb, which they developed, pressure tubes in their RBMK reactors and are the only country in the world that could possibly have information relevant to our pressure tube concerns. Further exchange meetings are planned with the USSR to establish the total extent of their knowledge with Zr-2.5 wt% Nb pressure tubes.

In 1989 the irradiation creep and growth program in the Osiris reactor at Saclay was started. This reactor provides a high fast neutron flux which will enable pressure tube materials to achieve end-of-life fluences towards the end of the 1990's. The irradiation growth program in Dido has been in place for about the last 12 years and has provided important information on the effect of microstructure and irradiation temperature on the irradiation growth of Zr-2.5 wt% Nb. The reason for the program in Osiris is that end-of-life fluence levels will be reached faster and at a lower cost than could be achieved with Dido. The current situation with Dido is that it is scheduled to be shutdown in April 1990. This is a result of a lack of customers for the facilities in Dido. Comparison of the initial Osiris growth data with that from Dido indicates that the irradiation conditions in Osiris will generate the same relationship of growth strain with neutron fluence as has been obtained in Dido.

The development of a technique for the nondestructive measurement of hydrogen concentration in Zr-2.5 wt% Nb continued in 1989. The technique reached a key milestone when it was shown to work on irradiated material. This has given a new impetus to the program with the next stage being the development of a unit to measure hydrogen concentrations in material removed from the Pickering 3 LSFCR. As part of the development of this technique, differential scanning calorimetry (DSC) has been used to give an independent measurement of hydrogen. Interest in this technique has been increased as a result of difficulties in obtaining accurate and reliable hydrogen concentration measurements by other processes. The DSC technique has the added advantage in that it does not destroy the specimen and so a number of repeat measurements can be made.

Decisions on the time when LSFCR of Pickering and Bruce reactors sh ld be performed have had input from the studies on hydride blister growth. From these studies a probabilistic code to predict blister formation in reactors where pressure tube/calandria tube contact occurs has been developed.

The work program on pressure tubes in the Materials Integrity Section will continue for several years. Pressure tubes are seen as the key components of the CANDU system which limit the capacity factors of the generating stations.

In support of thermal plants the corrosion/fatigue facility has been upgraded and has been shown to be able to produce service like cracking in the laboratory. Creep tests capabilities have also been improved and are available for tests on main steam line materials.

2.0 MATERIALS INTEGRITY SECTION PERSONNEL

R.G. Fleck	Supervising Engineer
L.J. Compton	Section Assistant
E.T.C. Ho	Engineer
S. Sandloehken	Technician - SEM
M. Leger	**Unit Head - Fracture Mechanisms**
W.K. Lee	Engineer
G.K. Shek	Engineer
K. Tashiro	Engineer
R.E. Irwin	Technician
O. Kiir	Technologist
D. Oad	Technician
H. Seahra	Technologist
P. Vesely	Technologist
R.A. Holt	**Unit Head - Deformation Mechanisms**
V. Perovic	Engineer
N. Badie	Engineer
A. Donner	Technologist
Z. Grozdanovski	Technologist
R. Jarochowicz	Technician
M.J. McGraw	Technologist
A. McMillan	Technician

Summer Students

Mrs. S. Perkons
Mr. T. Chong
Mr. V. Tychowsky
Mr. F. Lupo

Attached Staff

Consultant Dr. D.V. Leemans

Waterloo Work Term Student

Mr. R. Lecsek

3.0 FRACTURE MECHANISMS UNIT - 1989 REVIEW

3.1 Blisters

3.1.1 Crack Initiation and Growth from Hydride Blisters
(M. Leger, W.K. Lee, O. Kiir, P. Vesely)

The overall program objective is to develop an understanding of the mechanisms involved in the initiation, propagation and instability of cracks from blisters formed at the contacts between pressure tubes and calandria tubes. This understanding will be based upon analysis of experimental results of blister cracking tests and will be used to aid in the decisions which must be made with regard to operating pressure tubes in contact.

3.1.1.1 Blister Growth In Pressurized Tubes

Assembly of the equipment required to carry out tests under reactor thermal boundary conditions is nearing completion. Flow tests on acrylic models have allowed some optimization of the design. It appears now that we may be required to work with heat transfer conditions that are lower than those in reactor. This work has proceeded somewhat slower than anticipated due to a lack of available manpower. As a result, the original budget was significantly reduced.

3.1.1.2 Blister Stress Measurements

The hole drilling technique for residual stress measurement was applied to several blisters and to specimens of a tube used to grow blisters under stress. Results indicated that there were compressive stresses in the blister and that the magnitude of these stresses increased with blister size. The stress state at the blister surface became more tensile with increasing blister size. No residual stresses were found in the tube in which blisters were grown under stress. The trend towards higher fracture stresses in blisters grown under tensile stress may be a result of the final stress state in the blister being relatively independent of the stress state in the tube.

3.1.1.3 **Analysis of Previous Blister Growth Results**

A large effort in modelling of the blisters grown on deuterided tubes was undertaken. Sensitivity studies of key parameters of the model indicated that the parameters used as the reference case usually produced a good representation of the experimental results. There remains a need for completion of this work.

3.1.1.4 **Blister Review**

A review of all the work in the TRAC program was undertaken with a view to determining the key areas in which more work is required. This review is nearing completion. A report on the status of blister knowledge was presented at the fuel channel seminar.

3.1.2 **Probabilistic Modelling of Tube Fracture Due to Blisters for Reactor Assessment**
(M. Leger, D.V. Leemans)

The objective of this work is to develop and use a probabilistic computer model of pressure tube fracture due to blisters. This model would incorporate the results from the TRAC research program as well as information from the reactors on an individual basis to arrive at an assessment of the probable number of tubes likely to fail as a function of time.

The program designed to provide evaluations of two garterspring reactors were consolidated so that either piecewise linear deuterium pick up or data based deuterium pick up scenarios can be accommodated in a single program.

Work is currently underway to provide a similar program for CANDU 600 units which will incorporate a better contact time algorithm than that currently available. This will be applied to an assessment of Gentilly.

3.2 **Development of a Technique for the Nondestructive Evaluation of Hydrogen in Zirconium Pressure Tube Material**
(K. Tashiro and D. Oad)

The objective of this program is to develop a technique for the nondestructive evaluation of hydrogen (deuterium) in zirconium pressure tubes.

The technique being investigated relies on the detection of physical property changes as hydrides are being precipitated or dissolved in the Zr-2.5% Nb material. Hydrogen will influence the physical properties of Zr-2.5% Nb by producing a discontinuity in the temperature dependence of the property. Resistivity is the main property under investigation. Confirmation and calibration of the results obtained by these measurements will be confirmed by independent measurements obtained by calorimetry.

Calorimetry is capable of high resolution differential measurements by utilizing the heat of dissolution of hydrides. This factor was determined by relating the total heat absorbed by a sample to the hydrogen concentration of that sample. Preliminary calculation of lower detection levels have determined that approximately 0.4 ppm hydrogen concentration differences can be observed for 14 ppm samples. The differential concentration determined by this technique agrees well with that obtained by vacuum extraction methods as shown in Figure 1 and is likely to be more precise since nondestructive repeated measurements can be performed. This technique can also benefit hydrogen pickup studies where differences in concentration are more important.

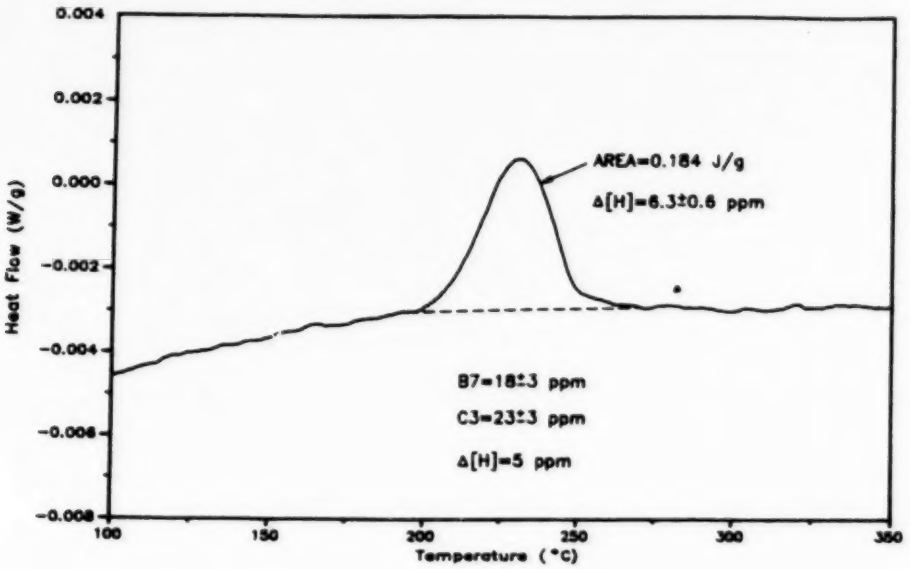


Figure 1 *High resolution calorimetry comparing samples B7 and C3. The vacuum extraction hydrogen concentrations used are the mean for the series. The calorimetric concentration difference agrees with the vacuum extraction concentration difference.*

Calorimetry is a technique which may provide indirect support by providing on site TSSd determinations of pressure tube scrapes or punches. Work is progressing to determine if nonirradiated scrapes and punches can be analysed. In addition, attempts are being made to isolate the system electronics from the main measuring cell so that irradiated sample work can begin.

The resistivity technique developed here requires minimal sample preparation. Preliminary tests on nonirradiated tube sections have shown that the pressure required to penetrate the oxide to make electrical contact is very low. However, at these low probe pressures, the contact resistance is unstable with increasing temperature resulting in an unstable signal output. A considerable increase in probe pressure is required to overcome this instability.

Tests with the new test rig have shown that the signal output stability and sensitivity is very sensitive to probe geometry and construction materials. Small changes to probe construction required to place these probes into the new test rig have resulted in unstable signal output and loss of sensitivity due to stray inductive effects.

Nonirradiated pressure tube sections have been tested with a hydrogen concentration range from 14 to 50 ppm. The collective nonirradiated TSSd results shown in Figure 2 are comparable with calorimetry and other methods.

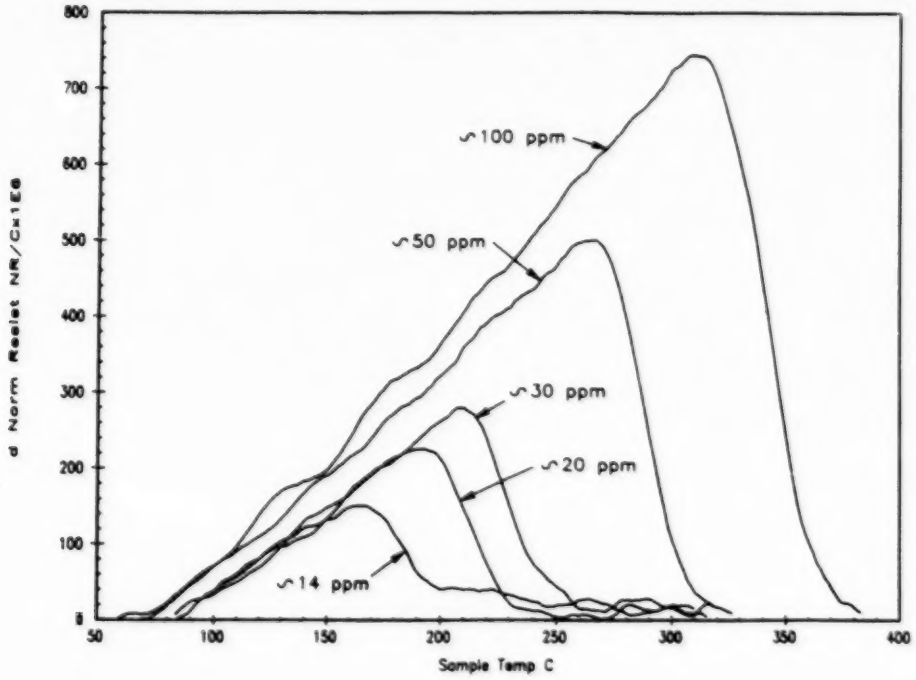


Figure 2 *Resistivity heating data composite illustrating hydride dissolution in nonirradiated pressure tube sections. The results are tube sections with five different hydrogen concentrations. Heating rate is 5°C/min.*

The collective nonirradiated TSSp results shown in Figure 3 are different from other experiments. This difference is attributed to the different cooling rates.

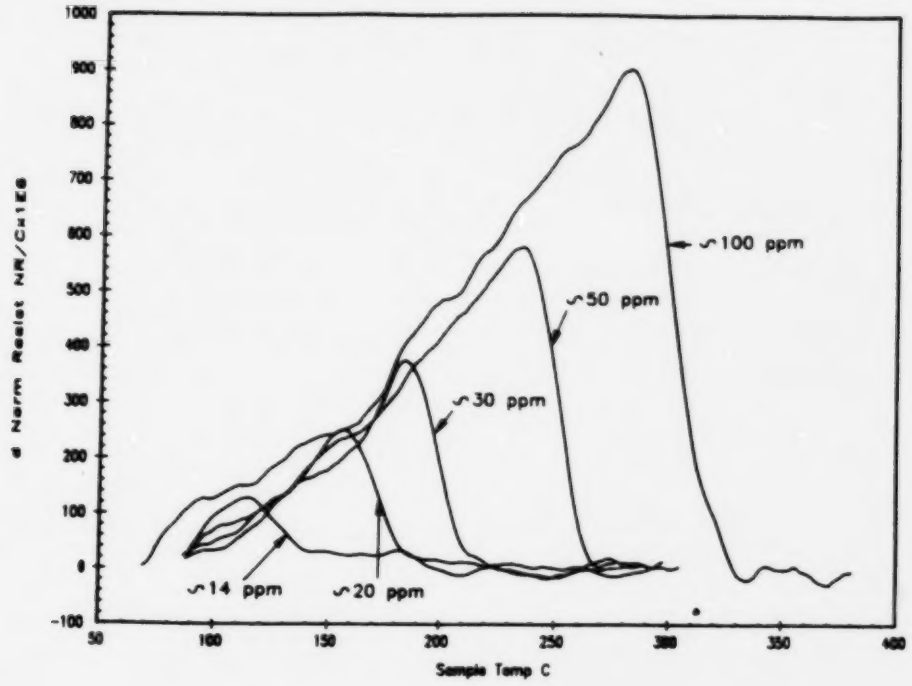


Figure 3 *Resistivity cooling data composite illustrating hydride precipitation in nonirradiated pressure tube sections. The results are tube sections with five different hydrogen concentrations. Cooling rate is 5°C/min.*

Since the sample consists only of a section of pressure tube, measurements on irradiated pressure tubes are easier to obtain. Preliminary tests on irradiated tube sections have also been successful. Specifically, the tubes tested were: B2U07 (1.57 m from the inlet), P3L09 (1.31 m from the inlet) and P4D19 (3.21 m from the inlet). Probe penetrations of the thicker oxide layer did not present any great difficulties at the time of measurement. These tube sections possessed both deuterium and hydrogen. The hydrogen equivalent concentration of these tube sections ranged from 14 to 30 ppm. The dissolution and precipitation peaks observed from the irradiated material were similar to the non-irradiated peaks. Both critical temperatures, TSSd shown in Figure 4 and TSSp shown in Figure 5, appear to be depressed with respect to nonirradiated tube sections. This was observed even after numerous heating and cooling cycles. These results may indicate complex dissolution and precipitation behaviour which differs from nonirradiated material. This may be a result of changes in microstructure due to irradiation or the presence of deuterium resulting in isotopic effects. This behavior requires further study.

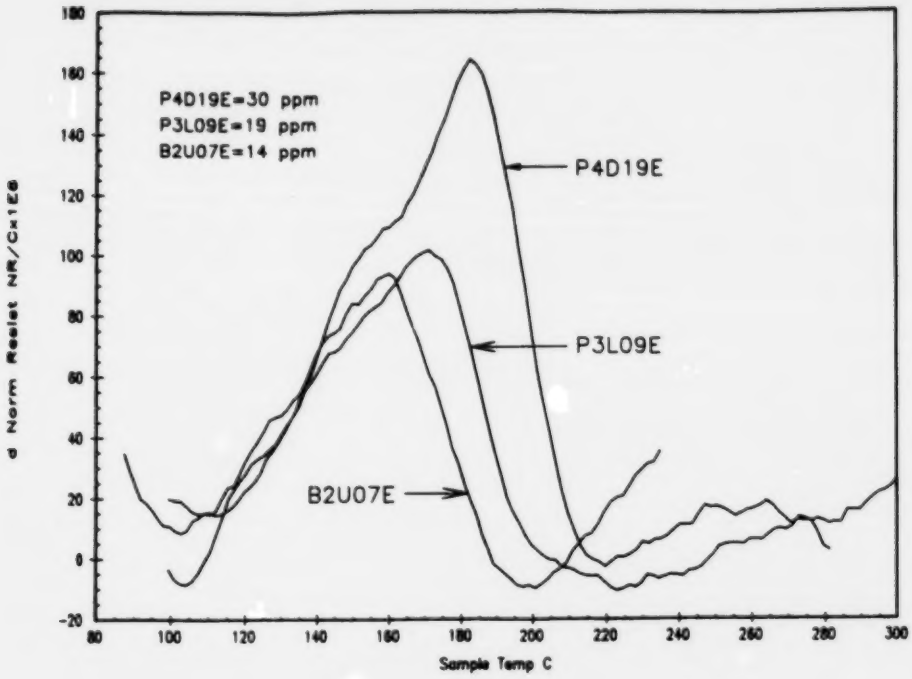


Figure 4 *Resistivity heating data composite illustrating hydride dissolution in irradiated pressure tube sections B2U07, P4D19 and P3L09. Heating rate is 5°C/min.*

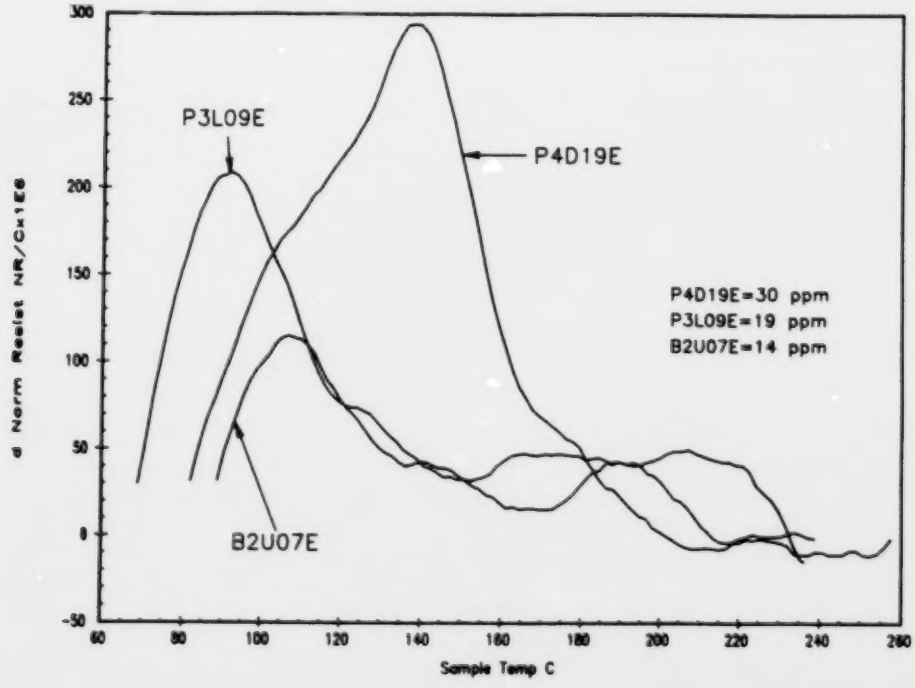


Figure 5 *Resistivity cooling data composite illustrating hydride precipitation in irradiated pressure tube sections B2U07, P4D19 and P3L09. Cooling rate is 5°C/min. Notice abnormal behaviour of P3L09 with respect to other tube sections.*

Overall, irradiated tube sections definitely exhibited the changes in resistivity due to hydride dissolution and precipitation. With the encouraging nonirradiated and irradiated tube section results from the resistivity tests, planning is now underway to test a larger number of irradiated tube sections which will be removed from the Pickering 3 reactor. Since the preliminary rig was not sufficiently reliable to test a large number of samples, another more robust rig is now being constructed. This rig will be based on an external heater system. Internal heating of the tube may present different problems, therefore a rig based on this heating system is also being constructed.

3.3 Delayed Hydride Crack Initiation

3.3.1 Rolled Joint Integrity

(P. Vesely and W.K. Lee)

This test program examines the behaviour of zero clearance rolled joints with machined defects under simulated reactor operating conditions. Rolled joints with 60 ppm hydrogen were pressurized with nitrogen to 10 MPa and thermally cycled between 300°C and 250°C to promote hydride precipitation.

To speed up the test program, the thermal cycling was changed from once to twice per week. The hold time at 250°C after each thermal cycle is more than two days; this time period should be sufficient for hydride precipitation due to the DHC mechanism. The total number of cycles has now reached 50, however, the acoustic emission monitoring system did not detect any cracking. There were only some AE events during each thermal cycle.

One set of rolled joints was removed from the furnace and will be sectioned for metallographic examination. After inspecting the other set of rolled joints, the thermal cycle experiment will continue.

3.3.2 Delayed Hydride Cracking and Fatigue

(P. Vesely, D.A. Scarth, M. McGraw, and W.K. Lee)

This project investigates the interaction between fatigue and delayed hydride crack initiation. Fatigue and constant load tests at 250°C were carried out on notched tension specimens with and without hydrogen.

The fatigue test results indicated that the cycles to crack initiation depended on the loading ratio R (minimum load/maximum load), the loading frequency and the presence of hydrogen. The observations are described below:

With a R ratio of 0.05, there was no significant difference between the hydrided material and the as-received material. Even when specimens with preformed radial hydrides were tested, there was no change in fatigue behaviour. The compressive stress field during the low stress part of the cycle prevented/stopped the DHC contribution. The damage mechanism was mechanical fatigue as opposed to DHC.

With a R ratio of 0.95, the hydrided specimens loaded to high peak nominal stresses have lower fatigue lives than those for the as-received specimens exposed to identical test conditions. Radial hydrides were found in these specimens. Furthermore, the observed dependency of the number of fatigue cycles on the test frequency suggested that the fatigue crack initiation process was influenced by a time dependent mechanism. When expressed in time to crack initiation, these specimens had about the same failure times as those in constant load DHC tests. However, there was less data scatter in the fatigue results. Figure 6 illustrates the data scatter at a peak elastic stress concentration of about 1900 MPa. The presence of cyclic loading helped to activate the onset of DHC crack initiation.

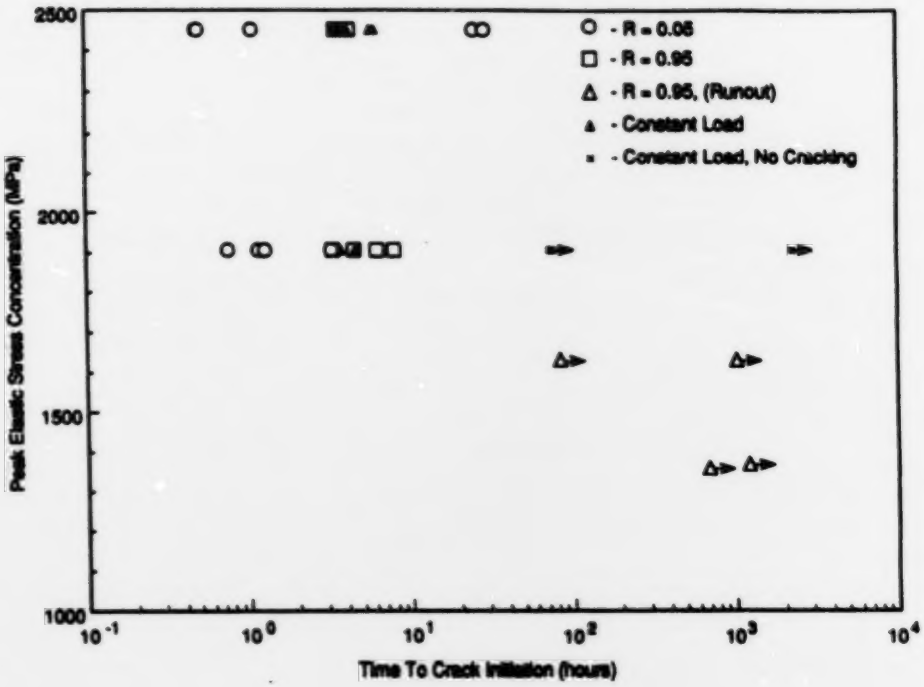


Figure 6 Time to crack initiation from hydrided Zr-2.5 wt% Nb notched tension specimens at 250°C

The fatigue damage parameter σ' which is the square root of (Young's modulus x peak effective stress x alternating effective strain) was used successfully to evaluate the fatigue data that included different loading R ratios. Neuber's rule coupled with deformation plasticity theory was used to determine the

notch-tip stress/strain values. Figure 7 shows all the fatigue results. Tensile tests at 250°C on as-received and hydrided material were also performed to generate material data for the above calculations.

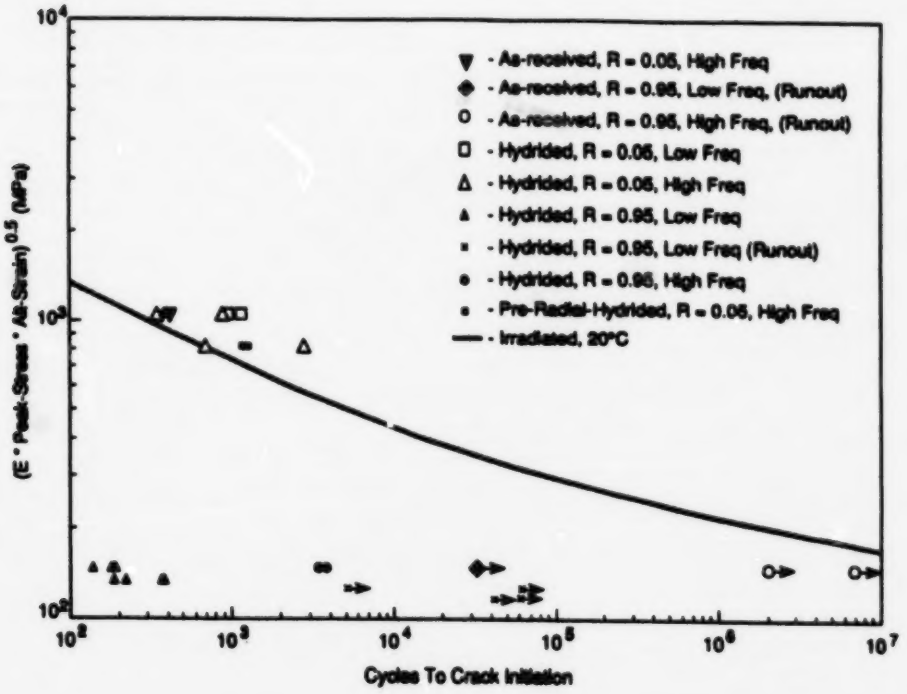

- As-received, R = 0.05, High Freq
- As-received, R = 0.95, Low Freq, (Runout)
- As-received, R = 0.95, High Freq, (Runout)
- Hydrided, R = 0.05, Low Freq
- Hydrided, R = 0.05, High Freq
- Hydrided, R = 0.95, Low Freq
- Hydrided, R = 0.95, Low Freq (Runout)
- Hydrided, R = 0.95, High Freq
- Pre-Radial-Hydrided, R = 0.05, High Freq
- Irradiated, 20°C
(E * Peak-Stress * Alt-Strain)^0.5 (MPa)
Cycles To Crack Initiation

Figure 7 Fatigue data points from as-received and hydrided Zr-2.5 wt% Nb notched tension specimens at 250°C. Also shown is room temperature curve from irradiated Zr-Nb.

3.3.3 Crack Initiation in Pressure Tube Materials

(P. Vesely, A.H. Chan, and W.K. Lee)

There were three main topics within this program: four point bend/cantilever beam crack initiation tests, probability model for crack initiation, and a parametric study on notched tension specimens with preformed radial hydrides.

Repeatable crack initiation times were obtained for duplicate four point bend specimens. Two specimens, each with a 0.8 mm deep notch and loaded to 400 MPa outer fibre stress, failed after 184 thermal cycles. This value is very

similar to previous test results. The ingress test results were also comparable to those from previous test samples. For the 0.75 mm notch/400 MPa specimens, cracking occurred after 5000 hours.

A surplus furnace was acquired and commissioned to increase our testing capacity. Loading rigs for cantilever beams were prepared and one is currently being tested. The clock system to record the failure of cantilever beam specimen was also designed and assembled.

A report on the probability model which relates the probability of cracking to the cycle time, applied stress, and notch depth was issued. We have also identified the need to investigate the role of isothermal hold and short cycle time on crack initiation.

Initial work to determine the criteria for crack initiation was successfully completed. The loading parameters to cause cracking in notched specimens with specific hydride distribution were measured. These data were then correlated with the notch-tip stress/strain data determined by finite element analysis. The onset of crack initiation at 250°C was detected by using dc potential drop. Preliminary results showed that similar peak stress values at the notch-tip, either hydrostatic or normal, were required to start cracking in both blunt and sharp notches. The peak stress location was comparable to the extent of radial hydride in front of the notch. There was also a distinct difference between the hydrided and the as-received material. The von Mises stresses did not show any correlation with the type of notch/the type of hydride distribution.

To improve the resolution of detection, an ac potential drop system has been selected and will be ready for testing next year.

3.4 Delayed Hydride Crack Growth

3.4.1 Crack Monitoring by Potential Drop

(T. Chong & G.K. Shek)

The objective of this exercise is to determine the feasibility of continuously monitoring DHC growth from an inside surface defect in a tube by dc potential drop with the probes placed on the outside surface. A 50 mm wide C-ring machined from a hydrided Zr-2.5 Nb pressure tube ring was used for this study. A semi-elliptical surface defect about 2 mm deep and 5 mm long was machined onto the inside surface of the C-ring. The ring was stressed to an outer fibre tensile stress of 400 MPa on the notched surface by increasing the diameter of the C-ring with a nut-and-rod arrangement. Delayed hydride cracking was initiated by heating the stressed sample to 300°C and then cooling to 250°C. Cracking was monitored by passing a direct current of 8 amps in the sample and monitoring the potential drop across the surface notch by two sets of probes, one on each surface. The results indicated that both sets of probes were able to detect crack growth and crack arrest, although the inside surface probe had higher sensitivity. Information obtained from this experiment will be applied to the biaxial crack tunnelling tests.

3.4.2 Hydrogen Ingress on DHC
(T. Chong & G.K. Shek)

DHC behaviour is greatly affected by thermal history. Laboratory experiments have shown that DHC velocity approaches zero in unirradiated Zr-2.5 Nb pressure tube materials at reactor operating temperature (250°C or higher) if the test temperature is approached by heating. It is postulated that the driving force provided by the difference in stress gradient modified TSS between the crack tip and matrix hydrides is insufficient to promote DHC. This driving force can be increased by hydrogen supersaturation which can be obtained by approaching the test temperature from above. Another way of increasing the amount of hydrogen in solution was by providing hydrogen ingress through the process of corrosion or electrolytic hydriding.

Electrolytic hydriding in a molten salt bath (KOH-NaOH) at ~300°C has been shown to supercharge Zr alloys with hydrogen above the TSS. In this experiment, a curved Zr-2.5 Nb specimen with a machined notch on the inside surface was precracked by DHC in a four point loading jig at 250°C in air. The precracked specimen was then placed inside a salt bath under load and hydrogen was charged into the specimen at 290°C with the temperature approached from below. Metallographic and fractographic examination was performed to evaluate crack extension that occurred in the salt bath. The test result showed that DHC appeared to be operative in the salt bath at 290°C with the test temperature approached from below. When the crack reached about 70% of the wall thickness, cracking was diverted from the radial-axial plane to the transverse-axial plane. This was probably due to the formation and cracking of large circumferential hydrides and the stress pattern through the wall in the four-point bend sample.

3.4.3 Crack Tunnelling
(G.K. Shek & H. Seahra)

A poster and a paper entitled "Crack Tunnelling in Zr-2.5 Nb Pressure Tubes by Delayed Hydride Cracking" was prepared for the Fuel Channel Seminar. The paper was a progress report on the work performed on tunnelling. The significant findings were that crack velocities in the axial and radial direction were comparable in the initial portion of crack growth from a surface notch. However the radial crack velocity was much reduced as the crack approached the opposite surface. DHC appeared to be operative in the thin ligament. Parametric studies showed that crack tunnelling increased when the crack grew up a temperature gradient through the wall with the temperature approached from below.

Another temperature gradient test was performed with a tensile specimen (3.8 x 50 mm gauge cross-section) in which the crack grew up a temperature gradient from 200°C to 250°C at an applied stress of 130 MPa. The test was terminated before wall penetration. Ultrasonic inspection showed that the crack had reached 28 mm with a profile similar to a previous temperature gradient test. The sample was cut in half along the centre. One half was broken open at

liquid nitrogen temperature for fractography and the other half prepared for metallography. The crack profile determined from the fracture surface confirmed the UT results. Metallography showed that a 0.14 mm wide ligament was left in the specimen. Radial hydride was observed at the crack tip. There was a hydrogen concentration gradient through the wall with hydrides accumulating in the cold contact region. However, there was still sufficient hydrides present in the hot side for DHC at test temperature. This test offered further evidence that tunnelling is encouraged in the presence of a temperature gradient through the wall.

A tunnelling experiment was performed to investigate the effects of stress relaxation in the ligament on DHC growth. In this experiment, a crack was allowed to grow from a surface notch at 250°C until a 0.5 mm ligament was left. DHC was stopped by raising the temperature to 320°C. This temperature was maintained for 200 hrs to allow for stress relaxation to occur in the ligament by creep. The temperature of the sample was then lowered to 250°C to provide the suitable thermal history for DHC. Test results showed that cracking was re-initiated right away when the temperature was lowered to 250°C. The re-distribution of stress during the 200 hrs hold at 320°C did not stop DHC in the ligament. Further investigation will be made using a wider tensile specimen and monitoring the stress in the ligament by strain gauging.

Ni plating was used to improve the edge retention in metallographic examination of fracture path in the thin ligament. The main objective was to seek evidence of DHC. Although the Ni plating greatly improved the edge retention, it was not possible to determine whether cracking in the ligament occurred along re-oriented hydrides.

3.5 EPRI Project No. RP1890-5: Corrosion Fatigue Boiler Tube Failures in Waterwalls and Economizers

(E.T.C. Ho and R.E. Irwin)

The main objective of this project is to develop a discriminatory laboratory test procedure whereby corrosion fatigue damage, representative of that observed in field failures, can be reliably generated under well-controlled test conditions in a section of boiler tubing.

3.5.1 Development of Equipment and Test Procedure

3.5.1.1 Flow Loop and Test Facility Developments

Development of the flow loop and test facility for mechanical cyclic compression loading of boiler tube specimens was completed early in the first quarter of 1989. Results obtained from commissioning tests carried out indicate that specified water chemistry conditions (eg, the B&W "contaminated" AVT water chemistry specifications) can be controlled and maintained under closed-loop flow conditions with a range of flow rates. The flow loop and test facility was ready for full tube testing under simulated service

conditions of elevated temperatures, high pressures and specified water chemistries by the second quarter. Late in the last quarter, the flow loop and test facility were upgraded by the additions of:

1) an electric pressurizing pump to replace the existing pneumatic pump which gave poor performance characteristics periodically;
2) a pneumatically-operated shut-off valve was installed immediately downstream of the preheater to isolate and shut down the flow loop in the event of specimen failure and/or a significant leak in the flow circuit;
3) a control circuit to stop load cycling when the pressure sensor in the flow loop is activated to shut down the flow loop.

3.5.1.2 **Test Procedure Developments**

Results obtained from the first full tube test carried out (FTT4) indicated that the experimental technique developed can produce ID-initiated corrosion fatigue cracks which are representative of those observed in field failures, under simulated-service conditions. However, during the course of tube testing, some refinements to the test procedure were made and they include the following:

1) the alignment of the load train assembly was improved by modifying the lock nuts which are used to attach the extension rods of the loading rig to the load cell and actuator ram of the MTS load frame;

2) a stress analysis of the current loading rig (MARK III) was carried out using a 3-D finite element code;

3) an ultrasonic crack detection system (Tokyo Keiki model SM90D) was acquired.

The ultrasonic unit was acquired to speed up the testing process. The system also has the potential for in-situ inspection of boiler tube specimens without having to remove them from the loading rig and flow loop.

3.5.1.3 **Experimental Tube Testing**

As mentioned above, the flow loop and test facility were ready for full tube testing early in the second quarter. The first full tube test (FTT4) was carried out using the following test conditions:

1)	temperature	-	~275°C tube ID surface temperature in the area of maximum stress concentration
2)	loop internal pressure	-	15.2 MPa (2200 psi)
3)	loop flow rate	-	10 litre per hour
4)	water chemistry	-	"contaminated" B&W AVT

5)	loading waveform and frequency	-	inverted haversine at 0.01 Hz
6)	cycle load limits	-	25 and 80 kN
7)	number of cycles	-	7000

The above test conditions were chosen because laboratory tests on crack initiation in tube specimens carried out by B&W indicated that the above temperature and water chemistry conditions were particularly conducive to the initiation of corrosion fatigue cracks. Detailed metallurgical examination of the above tube sample revealed that the ID surface subjected to maximum stress concentration had undergone accelerated oxidation and appeared much darker in colour than the unstressed areas of the tube ID surface. Within the darker colour patch, which was about 4 x 40 mm in size, a number of yellow-brownish pits could be seen. These pits were invariably associated with ID-initiated axial cracks. Transverse metallographic samples prepared from the cracked regions showed that the cracks were transgranular and oxide-filled. The depths of the cracks examined ranged from about 0.3 mm to about 3.0 mm.

A second full tube test (FTT5) was carried out using the same temperature, water chemistry, loop pressure and flow rate conditions as test FTT4. However, the peak compressive force was increased from 80 to 100 kN and the number of cycles was reduced from 7000 to 1000. Metallurgical analysis of the second sample revealed that a multitude of ID-initiated corrosion fatigue cracks were produced in the area of maximum stress concentration. In a given transverse section, the array of ID cracks was spread over an arc length of about 10 mm in the hoop or transverse direction of the tube. As shown in Figure 8, the ID cracks are typically transgranular and oxide-filled. The dominant cracks are 0.5-0.6 mm deep.

4.0 DEFORMATION MECHANISMS UNIT - 1989 REVIEW

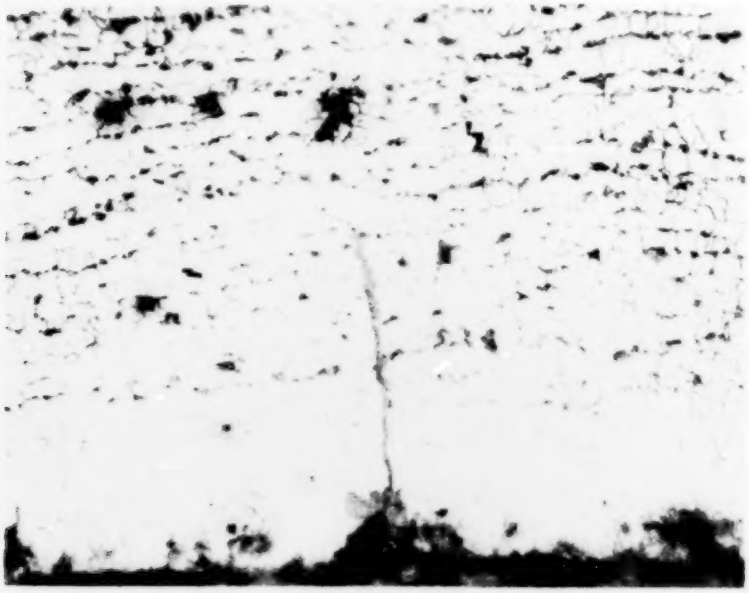
4.1 Reactor Data Analysis

4.1.1 CDEPTH Version 6

(N. Badie, R.G. Sauve and R. Lecsek)

Since the release of CDEPTH V6 in January 1989, its new capabilities and features have been tested thoroughly. A new revision of the code, V6.2, has been issued which contains several changes designed to make the code easier to use.

CDEPTH V6.2 has been ported to a DEC-3100 Unix workstation and several Intel 80386-based systems. The execution time for a typical CDEPTH run is of the order of 2-3 minutes on the DEC-3100 and 8-10 minutes on 80386 systems with the optimum hardware configuration. These compare with execution times of 30-50 minutes on a MivroVAX-II.



(a)

(b)

Figure 8 Micrographs illustrating the nature of ID-initiated corrosion fatigue cracks produced under simulated service conditions: (a) single crack nucleation locally; (b) multiple crack nucleation locally. (a) and (b) - 200X

With decreasing execution times for CDEPTH, preparing input files became considerably more time consuming and error-prone than running the program. For this reason, a user-friendly, menu-driven preprocessor has been developed for CDEPTH V6.2 in order to expedite the preparation of input files. The preprocessor, PRECD, is written in FORTRAN-77 and is operational on VAX/VMS, 80386/MS-DOS, and Unix systems.

4.1.2 Development of CRISPC
(R.G. Sauve and N. Badie)

A preliminary version of a finite-element program for simulation of fuel channel deformation using shell elements has been developed in cooperation with the Applied Mechanics Section. Because of their greater number of degrees of freedom, shell elements are expected to provide a more realistic simulation of fuel channel deformation than the beam elements used in CDEPTH. Both the static and creep solution routines of the program have been completed and tested. The creep routines of the program consist of an explicit creep solution driver, and a CRPLAW subroutine, which contains a two dimensional implementation of the CRNL-4000 and CRNL-4001 design equations, including the principal and in-plane shear anisotropy coefficients. Hence, CRISPC can model all the important modes of deformation of pressure and calandria tubes, i.e. sag, elongation and diametral expansion. Figure 9 illustrates a typical CRISPC model, showing the deflection profile of the pressure tube and calandria tube in a typical central zone channel in Bruce-2 (after 30 years at 80% C.F.).

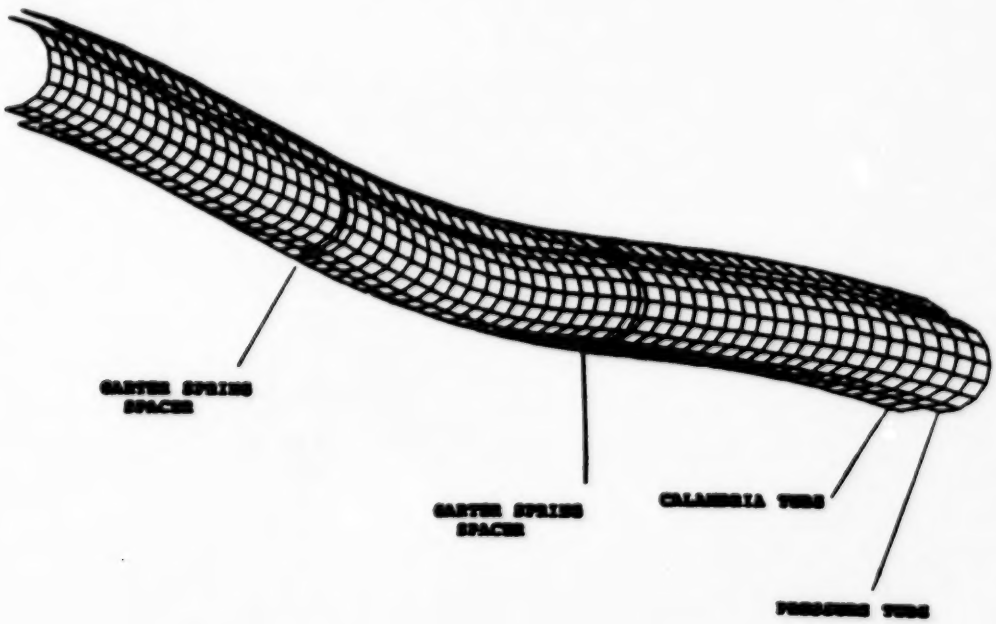


Figure 9 Fuel channel deflection profile Bruce Unit 2, high power channel

4.1.3 Fuel Channel Deformation Database

(N. Badie and S. Perkons)

A database has been developed using DBASE IV in order to make fuel channel deformation data more readily accessible. The database consists of several structured components, designed to contain different types of data:

(a) Operating Conditions Database (OCD): Each file in this database contains the time averaged fluence and temperature for all channels in a given reactor by bundle position. For convenience, measured elongation data are also included in this database.

(b) Diametral Expansion Database (DED): Files in this database contain the measurements and predictions of diametral expansion profiles at 40 to 50 points for channels that have been gauged.

(c) Sag Deflection Database (SDD): This database consists of files containing measured and predicted sag deflections, curvatures and slopes at 40 to 50 locations for pressure tubes and calandria tubes that have been gauged.

Thus, while the OCD files contain reactor data, files in the other two databases contain channel data. The data available at this time are listed in Table 1.

Table 1

Data Available in Fuel Channel Deformation Database

Database	Unit	Channel(s)	Date	EFPH
Oper. Cond.	P3	All	07/89	117300
	B2	All	12/86	61800
	B1	All	09/89	80800
Diam. Exp.	P3	J17,K05,M11,M12	1971	0
		K05,M11	1973	6800
		J17,M12	1974	13000
		J17,K05,M11,M12	1979	48600
		J17,K05,M11,M12	1982	72500
Sag	P3	J17,K05,M11,M12	1979	48600
		J17,K05,M11,M12	1982	72500
	B2	K07,L08,O13,T12	1983	35700
		K07,L08,O13,T12	1985	54500

4.1.4 **Contact Spreading**

4.1.4.1 **Design Equations**
(N. Badie and R.A.Holt)

The contact spreading database has been expanded to include the results of 1989 inspections in P4 and B2, as well as PIE results of the pressure tubes in B1-S04 and L1-K05. There are now 20 CIGAR results and 7 PIE results in the database.

The 7 PIE cases were used to test the design equations. The mean errors in the predictions of length of contact for these cases are -3 cm (minus sign indicates underprediction) for CRNL-4000/4001, and -17 cm for CRNL-1993/1956. Based on the standard deviations of the errors, the mean error of CRNL-1993 is statistically significant at 95% confidence level, while that of CRNL-4000 is not. A plot of calculated vs. measured lengths of contact in the PIE tubes is shown in Figure 10.

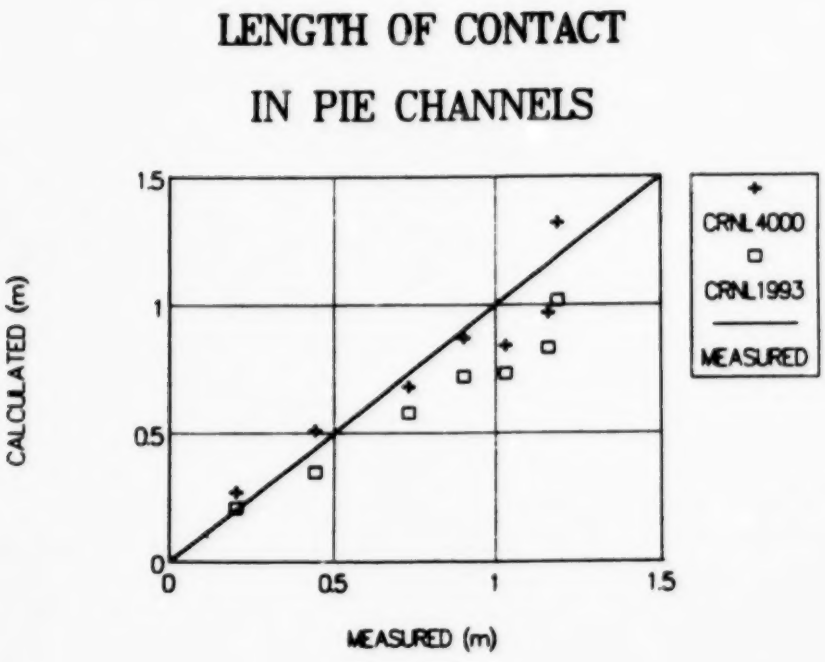


Figure 10

Predictions were also produced for time of initial contact and extent of contact for all P4 and B2 channels which were inspected. A comparison of these predictions with other contact predictions made for the same channels showed that the effect of uncertainties in the inputs was significant. In some cases,

differences in the inputs had a larger effect on the results than using different design equations, and could bias the assessment of design equations. Most of the differences in the inputs have now been resolved and a summary of agreed inputs will be issued.

4.1.4.2 **Contact Spreading Model**
(R.A. Holt, N. Badie and Z. Grozdanovski)

The current version of the contact spreading model uses times and locations of initial contact based on the CRNL-1993 design equation. If the CRNL-4000 design equation is used, different times and locations of initial contact will be obtained, resulting in different parameters for the model. The model has been refitted to the 9 contact measurements with initial contact data obtained from CRNL-4000.

In addition, a study of contact spreading has been carried out for low power and outer zone channels in Pickering and Bruce, and for units with thin spacers (5.59 mm coil diameter). For low power channels, the spreading rate varies linearly with the axially averaged channel flux. More work is required to quantify the effect of temperature; however, CDEPTH calculations show that temperature variations in Pickering and Bruce affect the average spreading rate in these reactors by less than 10% once the latter is scaled according to channel flux. For channels with thin (5.59 mm) spacers, the average rate of contact spreading towards spacers is 60% higher compared to units with 6.81 mm spacers under the same operating conditions.

4.1.5 **Analysis of Elongation Data**
(N. Badie, R.A. Holt and A. McMillan)

A study of elongation rates in Pickering-A and Bruce-A reactors was initiated, using the pressure tube elongation database developed by Ontario Hydro's Nuclear Engineering Department as the source of data. The purpose of this study is to isolate the effects of possible measurement errors, offsets and initial deformation transients on the elongation rate, so that a meaningful analysis of the effects of other parameters such as temperature, fluence, etc. may be carried out.

The results show that the elongation rates vary across the reactors in a manner inconsistent with the variation of operating conditions. Pickering Unit 4 data can be used to illustrate this effect: the data are plotted as average elongation rate vs. row number (1-22) and column number (1-A, 2-B, 22-W), as shown in Figure 11. The elongation vs. column number plot is roughly symmetrical, with dips at the ends. The elongation rate vs. row number plot, however, is asymmetrical, with the highest rate in row A, and the lowest in row W. These profiles cannot be caused by flux, temperature or fluence effects. One possible explanation is that some of the measurements are biased by distortion of the fueling machine plane. Further investigations of this and other effects are on-going.

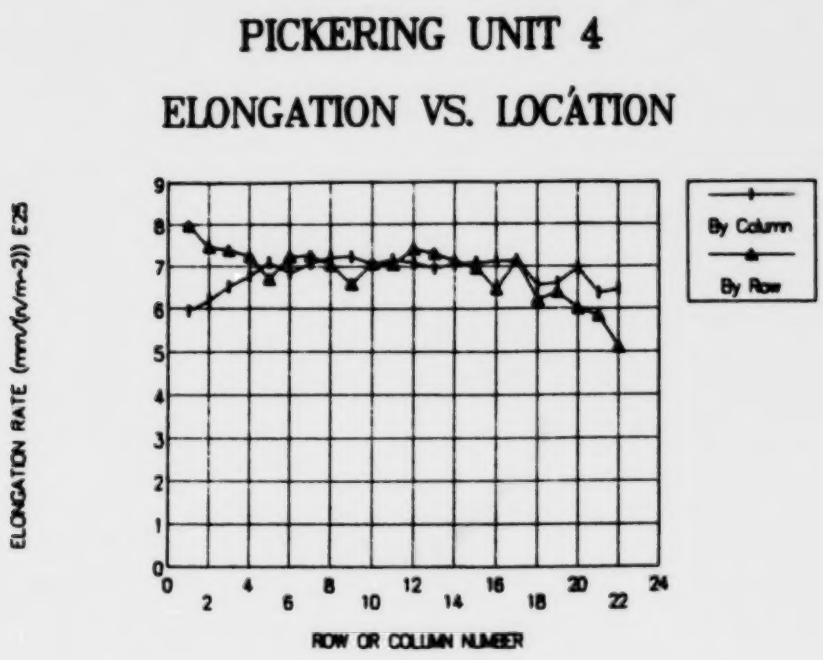


Figure 11

4.2 **Reactor Maintenance Support**
(N. Badie)

Various studies and short projects were undertaken in support of the fuel channel inspection and maintenance program. These include:

- determination of consistent flux and temperature profiles for use in various analyses of fuel channel deformation by designers.
- contact and sag predictions for various inspected channels.
- elongation, diametral expansion and wall thinning predictions to support evaluation of axial flaws (e.g. bearing pad scratches).

4.3 **Analysis of Stress Redistribution Near Circumferential Flaws**
(D.S. Scarth, D. Mok and N. Badie)

Because of the stress concentration near the tip of a notch, the displacement of material in this area due to creep is constrained by the surrounding material. This may lead to substantial stress relaxation near the notch tip with time. Such an effect could have major implications for our understanding of crack initiation in pressure tubes, and a key impact on the disposition of

observed flaws in reactors. For this reason, a study was initiated jointly with the Non-Destructive and Fracture Evaluation section to investigate this effect.

The objective of the initial part of the study was to demonstrate that stress redistribution occurs, and to estimate its magnitude. For this purpose, a two dimensional, plane strain finite-element model of a pressure tube cross-section with a U-notch was analyzed using ABAQUS. Since no creep law has been derived for the conditions which apply in this case (i.e. various combinations of high and low flux, high and low stress and high temperature), different combinations of existing creep laws were used to model the material behaviour.

Although problems with the creep law and the ABAQUS model hampered progress, the initial results indicate that creep stress redistribution may result in significant stress relaxation near the roots of flaws in pressure tubes. However, progress must be achieved in several areas in order to be able to quantify this effect: (1) a material model applicable to the conditions mentioned above must be derived, (2) a three dimensional finite element mesh must be used to properly model the constraints on local deformation, and (3) the experimental data on crack initiation must be re-interpreted in light of the results of this work.

4.4 Deformation

4.4.1 High Fluence Deformation of Pressure Tubes

(R.A. Holt, A. McMillan)

Osiris Program (Cooperative project with CRNL)

The irradiation of the creep and growth insert in Osiris, at a nominal temperature of 553 K, started in 1989/04. The insert is designated Trillium-2. On the second and third irradiation cycles, there were problems with temperature control with some specimens experiencing a short period at 580 K and some at 530 K. The third cycle was terminated after two days and the part of the insert containing the heater was replaced. For the completion of the third cycle the temperature was maintained at 553±15 K. It appears that temperature control of ±10 K can be achieved now that some experience has been gained in operating the insert.

The third cycle was completed in 1989/09, the specimens were measured during 1989/10 and the insert returned to the reactor 1989/11. The fluences obtained during the first phase were estimated by the CEA as 0.86 to 1.1×10^{25} n/m^2, E>1MeV, depending upon the location in the core.

The diameters and lengths of the seven pressurized capsules of cold-worked (CW) and stress-relieved (SR) Zr-2.5 Nb fuel cladding 10 mm in diameter x 46 mm long were measured using a dedicated rig built by the CEA as part of the program. The accuracy of strain measurement is about $\pm 2 \times 10^{-4}$ in diameter and 1×10^{-4} in length. The results after three cycles of irradiation are shown in

Figure 12. The strains are reasonably consistent with those observed in 15 mm diameter CW-SR Zr-2.5% Nb fuel cladding in NRU at 573 K, taking into account differences in temperature and dislocation density. The decrease in longitudinal strain with stress is due to the texture, which has a predominance of basal plane normals in the radial direction.

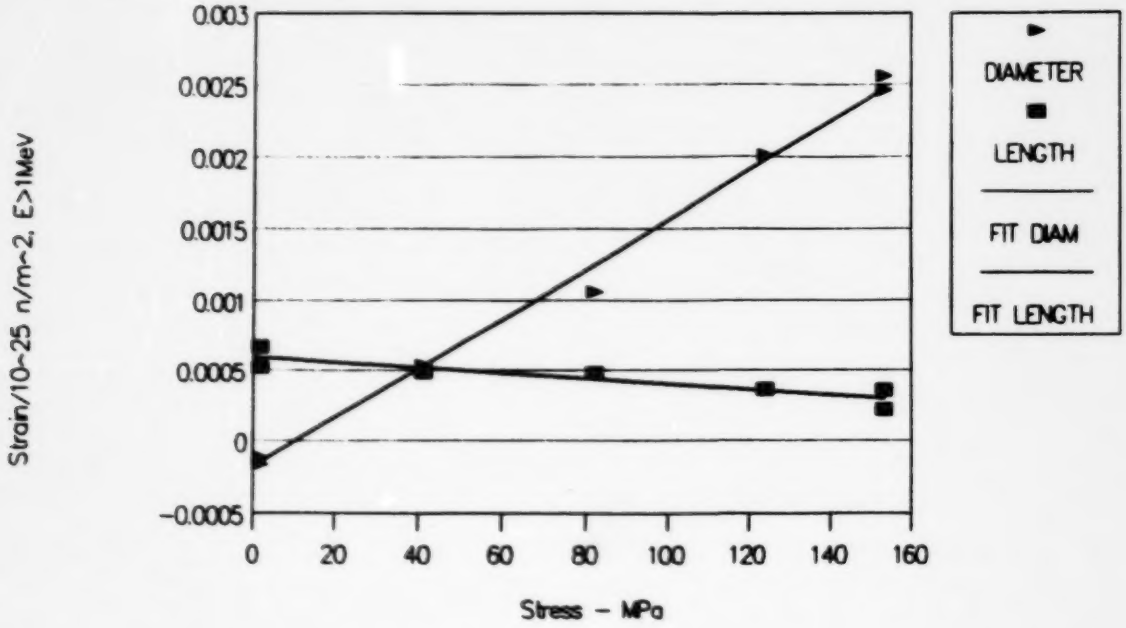


Figure 12 Creep data for pressurized Zr-2.5 Nb capsules from Phase 1 of the Trillium-2 experiment in Osiris.

The fuel cladding used for the creep capsules in this experiment was included after an unsuccessful contract with Nutech in 1988 to produce 10 mm diameter tubing by a route similar to that of pressure tubes. The manufacturing route for the cladding is typical of fuel cladding, and includes several stages of cold work with intermediate annealing. Based on the behaviour of the 15 mm diameter fuel cladding irradiated to moderate fluences in NRU, we felt the general trends of the in reactor behaviour would be applicable to pressure tubes.

However, during our visit to the Soviet Union in December 1989, we learned that Soviet growth experiments in the BOR 60 reactor have shown a difference in long term growth behaviour between CW and SR relieved Zr-2.5 Nb which had been prepared with and without intermediate annealing, Figure 13. The material without intermediate annealing which is probably representative of our

pressure tubes showed a much more drastic acceleration of growth at high fluences. The material with intermediate annealing which is probably similar to the fuel cladding used for the creep capsules showed less drastic acceleration.

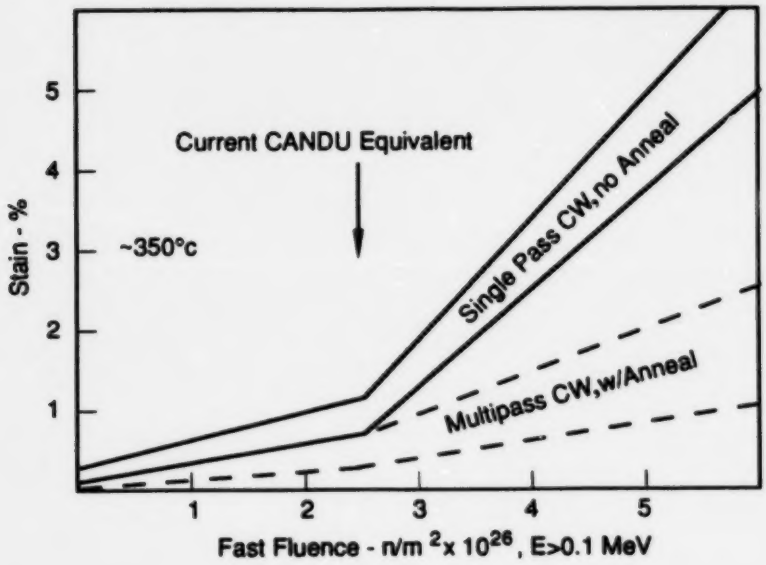


Figure 13 Growth data for CW and CW/SR Zr-2.5 Nb irradiated in the BOR 60 Reactor

A new contract with Nutech, started in 1989, has been successful to the extrusion stage in producing small diameter tubing more representative of pressure tubes. If the cold drawing proves successful, tubing may be available later in the year. In this case we will be recommending replacement of some or all of the original seven creep specimens.

The details of the 42 growth specimens are given in Table 2. Their lengths were measured using a dedicated rig of the type developed by AECL for the ATR growth program on calandria tube specimens. The accuracy of strain measurement is about $3x10^{-5}$. The longitudinal growth behaviour is shown in Figure 14. The strains of the standard pressure tube materials 1 and 3 - 7 were distributed about those exhibited by material 1 in DIDO, Figure 15. Material 1 exhibited somewhat higher strains than it had in DIDO averaging $4x10^{-4}$ in this fluence range in Osiris, vs about $2x10^{-4}$ in DIDO.

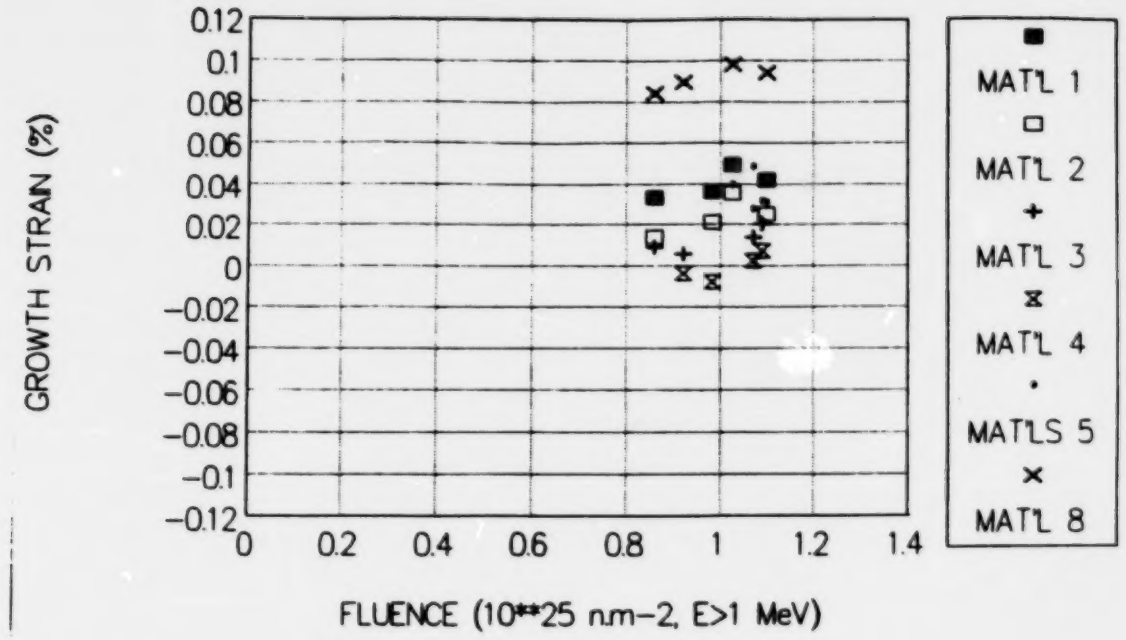


Figure 14 *Longitudinal growth data from Phase 1 of the Trillium-2 experiment in Osiris.*

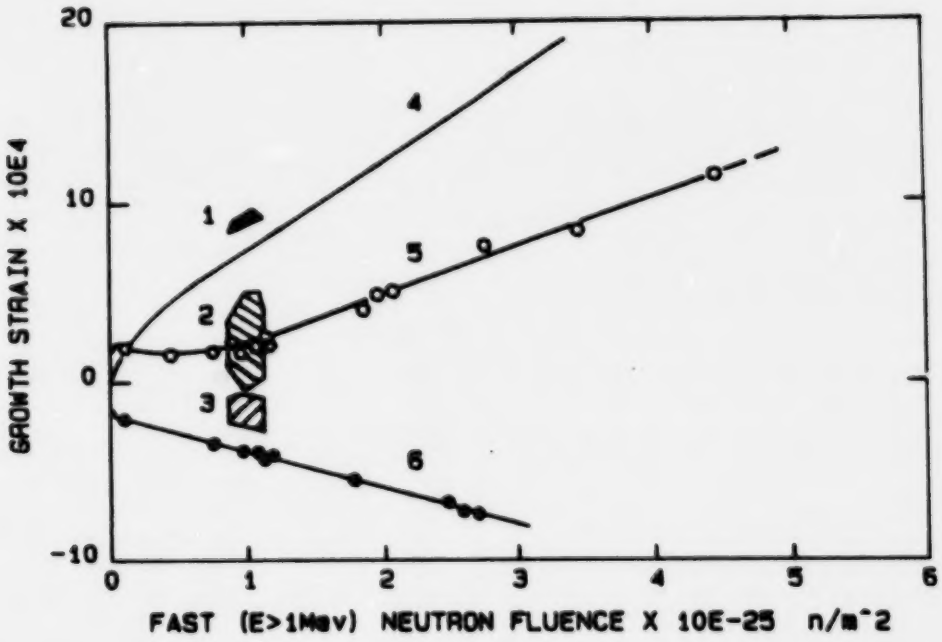


Figure 15 *Comparison of growth data from Phase 1 of the Trillium-2 experiment in Osiris (1 - Material 8, longitudinal; 2 - Materials 1,3 - 7, longitudinal; 3 - Materials 1,3, transverse) with growth behaviour previously established in DIDO (4 - Material 8, longitudinal; 5 Material 1, longitudinal; 6 - Material 1, transverse).*

Table 2

Schedule of Growth Specimens Being Irradiated in Osiris

#	Description	Tube #	Reactor	Numbers of Specimens			
				Initial Loading		Second Loading	
				L	T	L	T
1.	Recent production PT	H190	Darlington	4	4	4	3
2.	TG3 R1 " "	TG040	'88 Spares	4	4	4	3
3.	Intermediate " "	E064	Pt. LePreau	4	4	4	3
4.	Early " "	601	Pickering A	4	4	4	3
5-1.	360 ppm Fe B-quenched	D562	Pickering 8	1	-	1	-
5-2.	393 "	D866	Pickering 8	1	-	1	-
6-1.	1060 ppm Fe B-quenched	G710	Bruce 7	1	-	1	-
6-2.	1010 "	D1170	Pickering 7	1	-	1	-
7-1.	437 ppm Fe non B-quench	C363	Pickering 8	1	-	1	-
7-2.	663 ppm Fe "	D93	Pickering 6	1	-	1	-
8.	CW Zircaloy-2*	002	SGHW	4	-	3	-
9.	Preirradiated**	939	B2 P12	-	-	3	-
10.	Pilgered Tube	T007	-	-	-	1	1
Totals				26	16	29	13

* for calibration of fast flux. This material exhibits a high steady growth rate in DIDO and ATR and will be included in carrier bundles in Bruce A.

** preirradiated to ~7×10^{25} n/m^2, E>1MeV

The specimen to specimen variation for one material was less than the material to material variation within the narrow fluence range tested, ie $\pm 8 \times 10^{-5}$, vs $\pm 2 \times 10^{-4}$ respectively.

The reference material, cold worked SGHWR Zircaloy-2 pressure tube 002, material 8, chosen for its well characterized growth behaviour exhibited more growth in Osiris than in DIDO. This suggests that the Osiris produces more displacement damage per unit of fluence quoted. This might help to account for the higher strains observed in material 1. This will have to be confirmed by fluence measurements, not yet complete at Saclay, and further growth measurements.

The transverse growth behaviour of materials 1, 3 and 4 was different than that exhibited by material 1 in DIDO, the (negative) strains having a smaller magnitude, Figure 16. A possible explanation of this is that the behaviour of the transverse specimens irradiated in DIDO has been influenced by the necessity to flatten the material before stress relieving because of the longer gauge length (60 mm). The shorter Osiris specimens (38 mm) are machined from the wall of the pressure tube without flattening. The specimen to specimen variation was again less than the material to material variation, $\pm 3 \times 10^{-5}$ vs $\pm 7 \times 10^{-5}$.

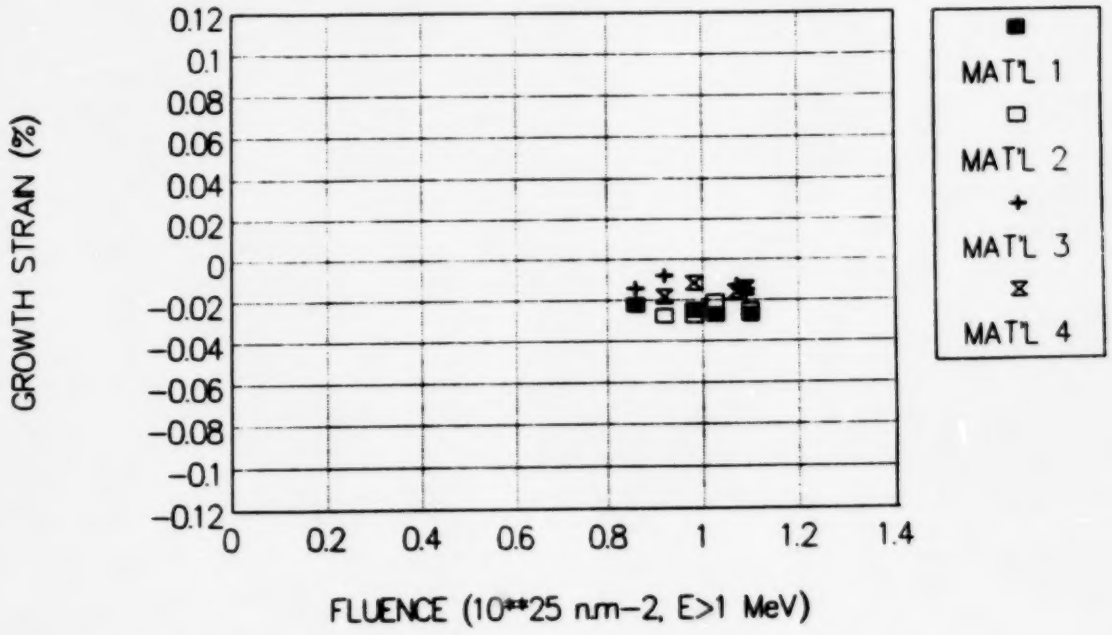


Figure 16 Transverse growth data from Phase 1 of the Trillium-2 experiment in Osiris

The growth behaviour of a 1988 production TG3 R1 pressure tube fell into the range of bahaviour of the standard pressure tubes and appears to be different from that of a 1980 pilot order tube observed in DIDO and in NRU. The average longitudinal strain in the fluence range was 2.5×10^{-4} vs 5×10^{-5} in DIDO and the average transverse strain was -2.4×10^{-4} vs -3.1×10^{-4}.

The fluence range is too small to determine the rate of growth accurately for any of the materials, and the observations so far may reflect predominantly the early transient behaviour. Further analysis of the results will await the next set of measurements so that the transient and steady state components of the strain can be assessed.

Because of the good reproducibility from specimen to specimen, it was judged that the earlier temperature variations in the insert were not of serious concern. Also, it was judged reasonable to reduce the amount of replication, and one each of the transverse specimens of Materials 1, 2, 3 & 4, and a longitudinal specimen of Material 8 were replaced. Three specimens from the surveillance pressure tube removed from B2 channel P12 in 1985 were included. These specimens have already experienced a fluence of about 7×10^{25} n/m^2, E>1MeV. They will remain ahead of the rest of the specimens in the program by about 6×10^{25} n/m^2, achieving a fluence of 2.1×10^{26} n/m^2, E>1MeV by the end of the first phase of the program (1995).

At the suggestion of OH CPS, longitudinal and transverse specimens of a production pilgered pressure tube were also included. Tubes of the latter type have been installed as replacements in Bruce 2, lattice position U07, and in Pt. Lepreau, lattice position K05. The new loading of the insert is shown in Table 2.

DIDO Program

Fast neutron flux measurements based on ^{93m}Nb have shown that the quoted neutron fluences for DIDO can be compared directly to those experienced by a CANDU pressure tube. TEM examination of growth specimens and of surveillance tubes showed similar microstructures, and the growth rates in Dido of surveillance tube material has been similar to that of growth specimens. These results suggest that the creep component of elongation is greater than previously believed as discussed in a paper presented at the 1989 Fuel Channel Technology Seminar.

Growth experiments were started this year on material with a fine (0.2 μm) and a coarse (0.5-1.0) μm grain size to test the model developed to explain the difference in behaviour of TG3 route 1 and standard pressure tube material. In the early stages of irradiation (1.6 to 4.1×10^{24} n/m^2, E>1MeV) the growth rates of the two specimens of the coarse grained material are positive while those of the two specimens of the fine grained material are negative - as predicted by the model. The same range of fluences was not tested in previous tests in DIDO so these growth rates cannot yet be compared with those of standard and TG3 pressure tube materials.

A specimen which has been irradiated for a fluence of 2.3×10^{25} n/m^2, E>1MeV, after being subjected to an increase in irradiation temperature to 623 K at 1.77×10^{25} n/m^2, E>1MeV and a decrease in irradiation temperature from 623 K to 553 K at 3.4×10^{25} n/m^2, E>1MeV, appears to be establishing a steady rate about three times higher than that exhibited by a similar specimen irradiated continuously at 553 K, Figure 17. Qualitatively similar behaviour is observed in a specimen irradiated at 583 K for a fluence of 1.77×10^{25} n/m^2, E>1MeV, after initial irradiation at a 623 K to a fluence of 1.62×10^{25} n/m^2, E>1MeV, but the ratio of rates is smaller. This clearly suggests the possibility of accelerated pressure tube elongation, both as a result both of nuclear maneuvering, and as a result of normal temperature variations associated with refueling.

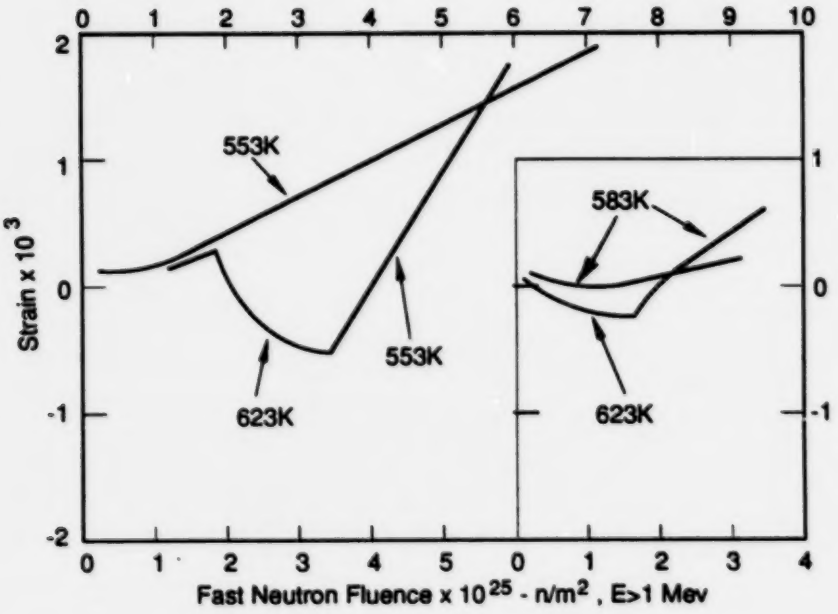


Figure 17 Effects of temperature change on the growth of Zr-2.5 Nb pressure tube material 1, in DIDO

4.4.2 Evaluation of Power Reactor Fast Fluxes
(R.A. Holt)

In this program creep, growth and mechanical test specimens of pressure tube materials are being irradiated in one or two special dummy elements in the outer element positions of carrier bundles. The test results will be compared with results of specimens irradiated in test reactors to verify the direct application of test reactor results to the power reactor environment. Displacement damage monitors and threshold transmutation monitors are also included to assess the quoted power reactor fluxes and flux spectra, currently based on theoretical calculations.

The first two bundles of the program were installed in Bruce Unit 4, channels M13 - position 2, and A10 - position 8, respectively, 1989/07/20. The third bundle was installed in channel Y11 - position 8, 1989/09/13. Irradiation of Bundle 2 will be complete in 1990/01.

The first bundle contained Nb and Fe threshold transmutation monitors, and sapphire displacement damage monitors was removed on 1989/08/30 and the carrier element was shipped to CRNL for disassembly. The monitors were successfully extracted and are being shipped to AEA Technology, Harwell for analysis. The carrier element of bundle 1, an outer element, received fast fluences, estimated from CPS Fueling histories to be 1.7 to 4.2×10^{19} n/m^2, E>1MeV at its outboard and inboard ends respectively.

The first phase of a subcontract with CANDU Operations and Zircatech to develop tools and procedures to assemble carrier bundles with irradiated specimens in the Pu fuel fabrication line at CRNL was completed and the hardware developed will be delivered to AECL early in 1990.

4.4.3 Advanced Carrier Bundles
(R.A. Holt)

A preliminary design has been completed by GE Canada of hardware to backload carrier bundles containing radioactive specimens from the spent fuel bay into a fueling machine through the irradiated fuel (I/F) port at Bruce or Darlington.

The dedicated mechanism replaces the standard air chamber, used to transfer irradiated fuel bundles from the fueling machine into the bay. The removal of the air chamber is a standard maintenance operation. The possibility of damage to the transfer port hardware is minimized by the use of a short adapter which is attached to the port before installing the backloading mechanism. All the standard interlocks are employed to prevent the mixing of the bay (light) water with the fueling machine (heavy) water. The backloading operation, including removal and replacement of the air chamber can be accomplished in approximately one shift.

The preliminary estimate of the cost of the system, including detailed engineering, drafting, procurement, manufacture and delivery to the station is $375,000.

4.5 Materials Developments

4.5.1 End-of-Life Properties of TG3 Materials

(R.A. Holt)

A contract with AECL/CRNL to prepare and characterize the specimens from a 1988 production TG3 Route 1 pressure tube was completed 1989/08 and the specimens were delivered to Saclay. The insert was loaded and irradiation commenced in Osiris 1989/09. After three cycles of irradiation, and a peak accumulated fluence of approximately 1.0×10^{25} n/m^2, E>1MeV, the first set of 18 specimens was replaced 1989/12 and are awaiting shipment to CRNL for testing. The temperature of irradiation during the first three cycles was approximately 520 K at the insert centerline, with estimated spatial variation along the length of the order of ±10 K - about twice that anticipated.

4.6 Tensile Properties of Recent Batches of TG3-R1 Pressure Tube Materials

(E.T.C. Ho and M.J. McGraw)

The test program undertaken to evaluate the strain rate sensitivity and related tensile properties of Zr-2.5 Nb pressure tube material with about 30% cold work (CW) was completed. This program was carried out to assess whether strain rate sensitivity played a part in the occurrence of ID surface shear cracks observed in some of the 1988 batch of TG3-R1 tubes. The two types of PT materials examined include a three cold-draw TG3-R1 material and an as CW Darlington-type material.

Results obtained indicate that both materials are strain rate sensitive over the ranges of strain rate and temperature studied. At the main test conditions of interest (50°C and 3×10^{-1} s^{-1}), the strain rate sensitivity index of both materials is about 0.011. This observation indicates that strain rate sensitivity did not play a significant part in the shear cracking. Analysis of the tensile data together with the available data for the original TG3-R1 pressure tubes indicate that the origin(s) of the observed shear cracking cannot be attributed to differences in bulk materials properties. However, the relatively high CW TG3-R1 material may be susceptible to shear cracking if the prevailing tube processing conditions are sufficiently different from those used previously for the original TG3-R1 tubes.

Tensile tests have been carried out on Zr-2.5 Nb PT materials in various conditions ranging from as-extruded to the fourth draw stage of the TG3-R1 fabrication schedule. The data obtained showed that both the 0.2% offset

yield strength and the tensile strength increase with increasing number of cold draw passes or cumulative CW. The increasing flow stress with CW has important implications on the cold drawing process:

1) other factors being equal, the drawing force increases with increasing material flow stress;

2) the ductility or the ability to accommodate additional plastic deformation before the onset of unstable plastic flow decreases with increasing flow stress.

From the above, it is evident that as the level of CW increases, the drawability of the Zr-2.5 Nb PT material decreases.

4.7 **Pressure Tube Fabrication Processes**
(E.T.C. Ho)

A computer literature search was conducted on the general topic of metal forming. The principal objective of the search was to obtain information which may be of relevance to the fabrication of Zr-2.5 Nb pressure tubes. The results obtained indicate that the number of publications of direct relevance to PT fabrication is very limited. However, the large number of papers published on the deformation behaviour of hcp metals and alloys under various metal forming conditions should prove useful in the analysis of pressure tube deformation behaviour under cold drawing conditions.

A contract research program funded by the Materials Integrity Section was awarded to Professor Zhurui Wang of the Department of Metallurgy and Materials Science of the University of Toronto. The main objectives of this research contract are:

1) generate some of the basic materials data needed to develop a more fundamental understanding of the deformation behaviour of the Zr-2.5 Nb alloy during metal working processes relevant to pressure tube fabrication;

2) increase the very limited number of researchers conversant with the deformation behaviour and metal working characteristics of Zr-2.5 Nb PT material;

3) develop finite element stress/strain analysis codes suitable for use in analyzing the deformation behaviour of Zr-2.5 Nb pressure tubes during metal working processes relevant to PT fabrication.

4.8 Materials Properties and Biaxial Deformation Behaviour for CANDU Reactor Zircaloy-4 Fuel Cladding at Elevated Temperatures in Air and Steam Environments (COG WP2887)

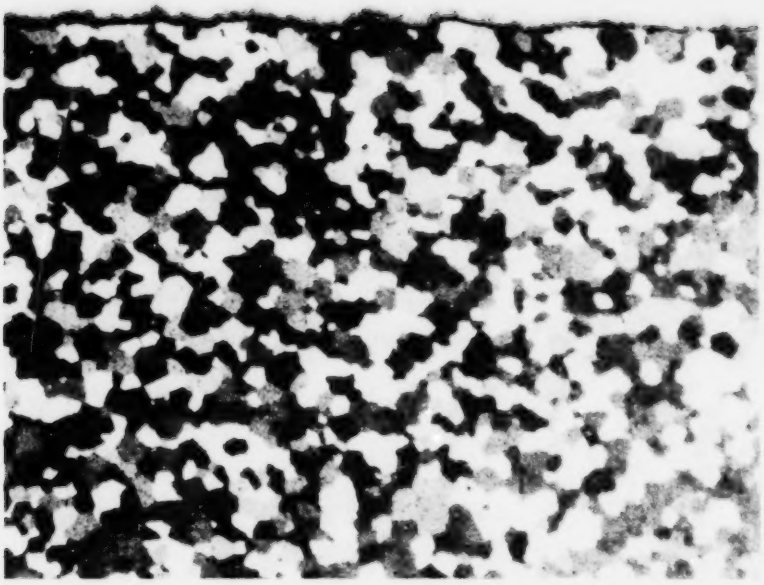
(E.T.C. Ho and A. Donner)

4.8.1 Materials Properties for Steam Oxidized Parent and Braze HAZ Zr-4 Cladding

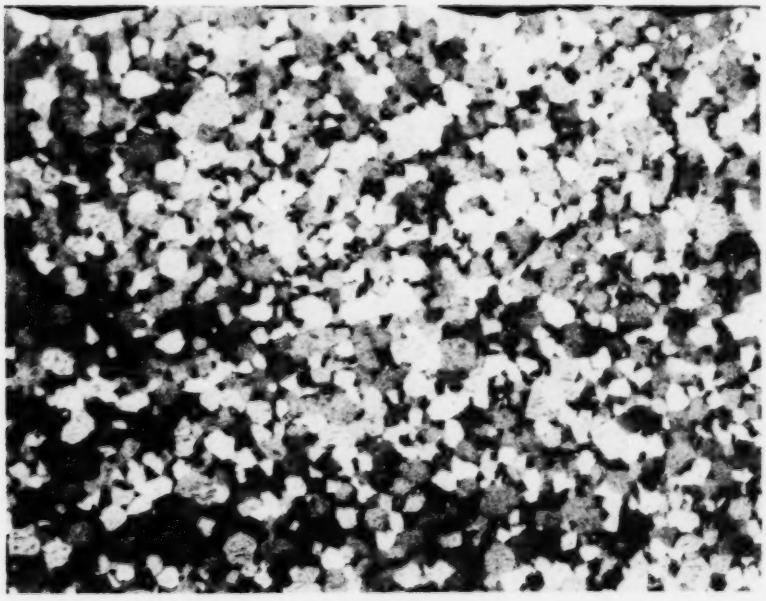
The test program to generate materials data for steam oxidized Zr-4 fuel cladding material similar to those obtained previously for air oxidized samples was completed. A total of 196 miniature D-ring tensile tests was carried out. Results obtained indicate that the Zr-4 samples oxidized in steam were much more severely embrittled than comparable samples oxidized in air. Metallographic examination of air and steam oxidized parent material samples indicate that the greater embrittlement associated with steam oxidation is related to the uptake of hydrogen by the substrate metal during the high temperature exposure. For example, the micrographs presented in Figure 18 indicate that for an exposure of 24 hrs at 650°C, the sample oxidized in steam absorbed sufficient hydrogen to transform some of the α-Zr grains to β-Zr phase which appeared as transformed β-grains upon water quenching to ambient temperatures in the micrograph in Figure 18(b).

4.8.2 Biaxial Deformation Behaviour in Steam

A large number of biaxial deformation tests in a steam environment was carried out successfully. The results obtained indicate that under comparable conditions of differential internal pressure and hold temperature, the biaxial deformation behaviour in steam is quite different from that observed previously in an air environment. A comparison of the biaxial tests carried out in air and steam at 650°C is presented in Table 3 and Figure 19. The data show clearly that, except at very low differential internal pressures, the time to rupture is much more rapid in steam than in air. Preliminary analysis of available data indicates that this difference in biaxial deformation behaviour at elevated temperatures is related to the uptake of hydrogen produced by the zirconium-steam oxidation reaction. However, at very low differential internal pressures, where the principal degradation mechanism appears to be the reduction of substrate metal wall thickness due to oxidation, the sample deformed in a steam environment lasts slightly longer than that in air, as shown in Figure 19. The above observation is consistent with the oxidation kinetic studies which showed that under all test conditions studied to date, the oxidation of both parent and braze HAZ materials are slower in steam than in air.



(a)



(b)

Figure 18 Micrographs illustrating the nature of the microstructure of parent cladding material oxidized at 650 C for 24 hours followed by water quenching to ambient temperatures: (a) in an air environment; (b) in a flowing steam environment. (a) and (b) - 400X

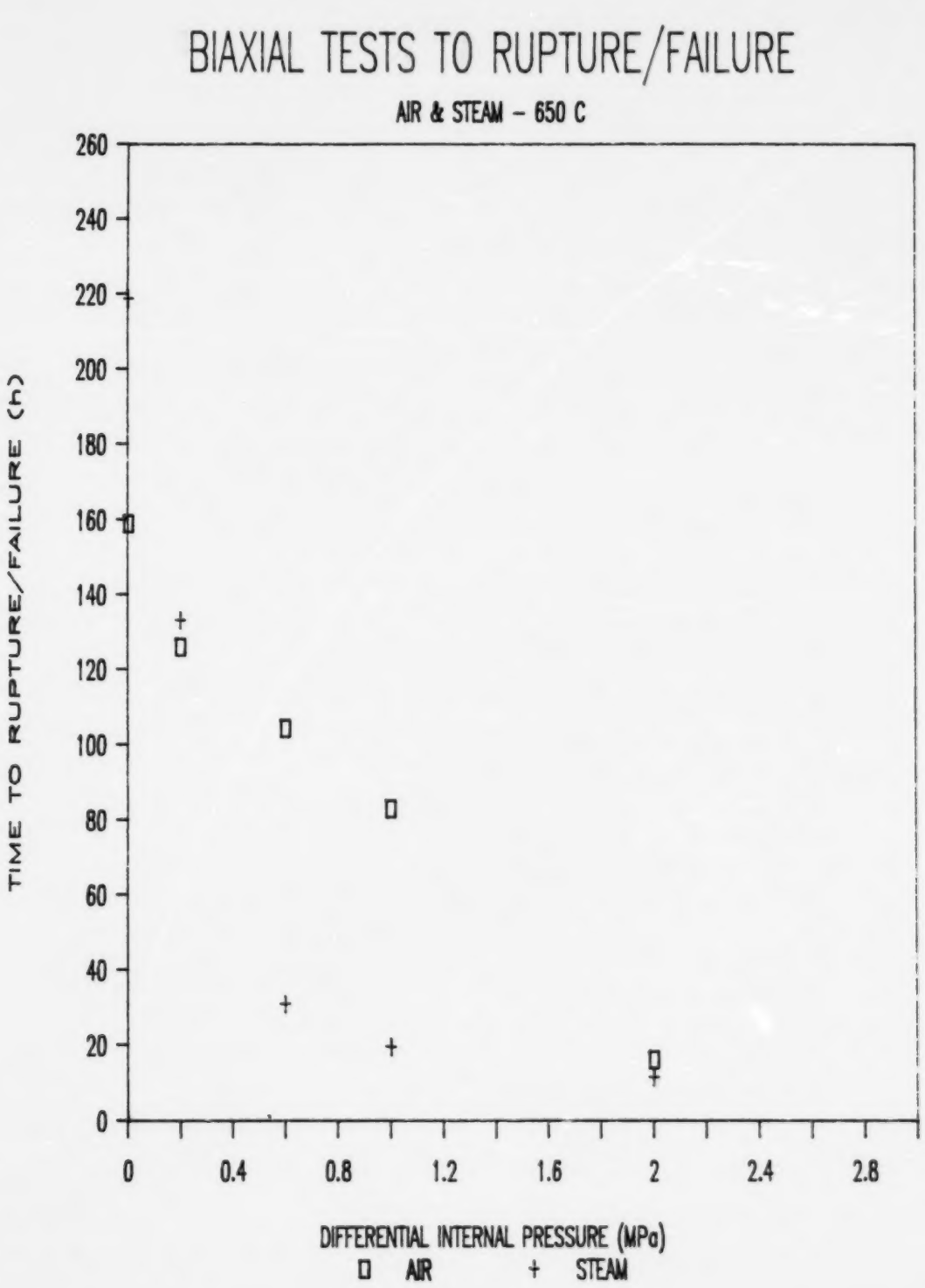


Figure 19 Plots illustrating the time to rupture/failure as a function of differential internal pressure in Bruce-type Zr-4 fuel cladding samples deformed under biaxial conditions at 650 °C in air and flowing steam environments.

TABLE 3
Summary of Biaxial Tests at 650°C

ΔP (MPa)	TIME TO FAILURE/RUPTURE (h) AIR	STEAM
0.2	125.9	133.1
0.6	104.3	30.8
1.0	77.9	19.5
1.0	80.9	
1.0	90.0	
2.0	16.1	11.4

4.8.3 Induction Heating System

The induction heating system for heating Zr-4 samples rapidly to temperatures in excess of 1000°C has been upgraded by the acquisitions of a new power transformer and an additional capacitor. This system has also been tuned to give optimal heating of full length Bruce-type Zr-4 fuel cladding samples. Preliminary heating tests conducted indicate that the enhanced system is capable of heating full lengths of Bruce-type Zr-4 specimens from ambient to elevated temperatures in excess of 1000°C at heating rates up to about 30°C/s. The above heating capabilities are needed to carry out the following types of experiments:

1) oxidation kinetic and biaxial deformation tests in air and steam for test temperatures greater than about 800°C;
2) experiments to assess the Be-braze assisted cracking mechanism in steam for temperatures in excess of 1000°C.

For test temperatures above about 800°C, it is crucial that the heat up time be a small fraction of the hold time at temperature. This is difficult or impossible to achieve using resistance furnace heating.

4.9 Materials Testing Using the Computer-Controlled Materials Characterization Test Facility
(E.T.C. Ho and M.J. McGraw)

4.9.1 Increased Test Capabilities

In the past year, the capabilities of the computer-controlled Materials Characterization Test Facility (MCTF) was enhanced significantly by the acquisitions of an AT-type personal computer and a Microstar model DAP 1200/4

data acquisition processor. These two pieces of equipment enable us to set up a data acquisition system parallel to that resident in the MTS computer control package. The main advantages of the new PC-based system are the greater speed available (data rates up to about 150,000/s) and the compatibility with other computer systems in terms of transfers of data and data analysis programs. Most data analysis programs prepared previously for use with the MTS DEC 11/23 computer system resident in the MTS computer control package have been modified for use with the PC-based system.

4.9.2 Method to Produce ID-initiated Cracks in Zr-2.5 Wt% Nb Pressure Tube Samples

The procedure developed previously to produce sharp ID-initiated fatigue cracks in a section of Zr-2.5 Nb pressure tube was tested in detail and refined in the past year. The method developed has a great advantage over other methods in current use in that it does not require the use of a machined flaw (ie, starter notch) for stress concentration to initiate a crack. Results obtained to date indicate that axial ID-initiated cracks can be produced in sections of Zr-2.5 Nb PT samples ranging in lengths from 10.2 cm (4") to 38.1 cm (15"). However, the experimental technique still requires some refinement before a single axial, ID-initiated crack of a specified depth can be produced in a reliable and consistent manner.

At present, there are two immediate applications for the above procedure and they are as follows:

1) production of tube samples with a range of known crack depths for calibration purposes in the ultrasonic (UT) inspection of reactor fuel channels;

2) production of tube samples with sharp ID-initiated cracks with prespecified depths for use in crack growth and other fracture mechanics-related studies of pressure tubes under internal pressure loading conditions.

4.10 Creep Testing Laboratory

(E.T.C. Ho and M.J. McGraw)

4.10.1 Increased Test Capabilities

All six (6) creep machines in the creep testing laboratory have been refitted with digital set-point temperature controllers. The use of these controllers for the three zone resistance furnaces for the creep machines increases the efficiency and accuracy of testing by facilitating more accurate and rapid attainment of desired test temperatures. In addition, one of the creep machines has been fitted with temperature controllers which can perform temperature cycling. This machine is also equipped with a constant stress lever and is intended for use with creep testing of Zr-2.5 Nb pressure tube samples.

Two high temperature extensometers (JVK extensometers) were also acquired for use with the two creep machines designated for creep testing under constant stress conditions. These extensometers will allow accurate measurements of sample strain during creep testing and they are intended for use with creep testing of Zr-2.5 Nb pressure tube material.

4.10.2 **Creep Testing of Thermal Plant Steam Line Materials**

The discovery of type IV creep cracking in the main steam line at Lakeview has resulted in an extensive plant investigation. In support of this plant investigation, a small test program has been carried out to assess the creep properties of steam line weld material removed from Lakeview Unit 6.

The data obtained from the above test program showed clearly that the failures occurred exclusively in parent piping or valve casting materials outside the weld heat-affected zone (HAZ). These observations indicate that the type of creep damage observed in the failed components has not been reproduced in the accelerated creep tests. However, since only a small fraction (~2 mm) of the nominal gauge length (25.4 mm) of the creep specimens contained the microstructure of interest (ie, the fine-grained region in the weld HAZ between the weld and the parent piping or valve casting materials), it appears that the initial creep specimen geometry chosen needs modification.

In view of the above observations, two creep specimens were machined with 1 and 2 mm root radius notches in the fine-grained region located approximately in the middle of the nominal gauge length of the specimen. Both samples were machined with the weld/casting interface in the middle of the nominal gauge length. These two specimens were tested at 657°C using a net section stress of 50 MPa. The above conditions were selected because these conditions produced a failure location very close to the fine-grained region in a regular (ie, unnotched) sample in the initial test program.

Results obtained from these two notched specimens indicate that the creep properties of the fine-grained region is superior (ie, longer times to failure) to that of the valve casting material; at least when the fine-grained material exists as a narrow band sandwiched between weld and valve casting zones. The above observations indicate that the quantitative evaluation of the materials properties of small zones of approximately uniform microstructures, such as those present in a weld, is very difficult. It appears that the best approach would be to produce larger samples, with microstructures similar to those of interest, which can be tested in detail. In principle, simulated weld microstructures can be produced by heating systems such as:

1) the commercially-available Gleeble system;
2) programmable AC self-resistance heating;
3) programmable induction heating.

The systems described under items 2 and 3 are available in the Materials Integrity Section's high temperature deformation laboratory.

5.0 ELECTRON OPTICS

Work continued on both the transmission and scanning electron microscopes through 1989.

The TEM studies concentrated on the evaluation of microstructures produced as a result of exposure of Zr-2.5 Nb pressure tube materials to fast E > 1 MeV neutron fluences. Materials from surveillance pressure tubes and from Dido growth specimens were examined. To increase the TEM capabilities, Dr. V. Perovic has been on attachment to the Cavendish Laboratories at Cambridge University. At these laboratories Dr. Perovic has access to state-of-the-art analytical transmission microscopes and can also resolve the fine details of the irradiation damage. This type of work is not possible with the old TEM in OHRD. Dr. Perovics' attachment will end in August 1990 after one year at the Cavendish Lab.

In the SEM area, the 840A instrument has been extensively used on a Divisional level. The main use of the instrument was in support of long-term research programs primarily related to corrosion problems in steam generator and heat exchanger tubes.

1990 PROGRAMS

6.0 FRACTURE MECHANISMS UNIT

6.1 Blisters

1. Blister growth and fracture on irradiated material. Cantilever beam rigs are being assembled for use in the hot cells at WNRE. Blisters will be grown on irradiated material under stress using these rigs. DHC will be initiated and the conditions leading to DHC from blisters in irradiated material will be studied.

2. Completion of the blister review including identification of any new work required. A report is in preparation.

3. Completion of blister growth model comparison with experiment and production of a summary report. Modelling of thermally cycled blisters needs attention.

4. Completion of the rig for growing blisters in pressurized tubes. Carrying out two initial tests of the blister growth to DHC and failure.

5. Initiation of some interaction with Universities on blister fracture. Current plans are to try to get fracture experts at McMaster involved in the program if possible.

6. More measurements of the blister stress states using the hole drilling technique.

6.2 Crack Initiation

To understand the crack initiation process in order to provide accurate prediction of pressure tube performance for operations in which delayed hydride cracking may be important.

1. The correlation between the critical stress state parameters that cause crack initiation and the type of hydride distribution at the notch-tip will continue. An improved technique to detect the onset of cracking will be examined.

2. The development of test procedures to obtain the cycle stress/strain properties will improve the fatigue analysis carried out on the notched specimens.

3. In addition to the four point bend and cantilever beam tests to produce crack initiation time data, long-term creep relaxation behaviour will be examined. The crack closure phenomenon will also be investigated with respect to the threshold for crack initiation.

4. Metallography on thermally cracked zero clearance rolled joints will determine the hydride distribution near the machined notches.

6.3 Crack Growth

6.3.1 Crack Tunneling

Parametric studies on crack tunneling will be continued using large tensile specimens with 50 mm wide gauge section. The wide gauge section allows longer axial crack growth by potential drop, strain gauges will be used to detect changes in strain in the samples as a function of crack depth. Only a small number of tests have been done so far to examine the effects of applied stress, temperature and temperature gradient on crack tunneling. More tests will be performed to assess their effects on tunneling.

Results obtained so far indicated that DHC might be operative in the thin ligaments but at a much reduced velocity. The cracking mechanisms in the ligament needs to be confirmed by metallography and fractography. The slow-down of crack growth in the thin ligament could be due to a reduction in driving force for hydrogen migration to the crack tip and loss of stress triaxiality required to crack the hydrides. The driving force for hydrogen migration and critical conditions for cracking hydrides could be affected by hydrogen concentration in solution, stress relaxation and material strength. Experiments will be performed to determine the effects of these parameters on crack tunneling using large tensile specimens.

It is also important to determine the critical conditions for wall penetration. The crack mouth opening displacement and the stress distribution in the remaining load bearing area at wall penetration will be estimated by metallography and strain gauging. The results will be compared with the modelling work by E. Smith as well as Scarth and Vanderglas (WPIR 6534).

Parameters found to have significant effects on tunneling in the tensile specimen program will be further investigated under biaxial loading conditions in a pressurized tube.

7.0 DEFORMATION MECHANISMS UNIT

7.1 Analysis of In-Service Deformation Behaviour of Fuel Channels

The objective of this program is to understand deformation behaviour of current fuel channels in power reactors, to provide support for fuel channel maintenance programs and provide input into design equations for pressure and calandria tubes and to develop analytical tolls and numerical models to study the deformation of fuel channels.

Although the phenomenology of pressure tube deformation is broadly understood, assessments of the elongation, sag and diametral strains of pressure tubes in Pickering 4 and Bruce 2 have shown tube to tube variations which do not correlate with operating conditions. A pronounced end-to-end effect is observed in the diametral behaviour which is not understood. Furthermore, the deformation behaviour cannot be fully rationalized with irradiation growth results from test reactors.

The phenomenology of calandria tube deformation is also broadly understood, but the models for calandria tube deformation based on small specimen tests have not been fully confirmed by reactor measurements.

Comparisons of power reactor dimensional data with model predictions are sensitive to some operating conditions and consistent values from one reactor to the next are not always available. Since the equations used to predict fuel channel deformation are normalized to power reactor dimensional data, it is necessary to have a well documented data base for both dimensional data and operating conditions.

The importance of these considerations is all the more clear with the necessity to assess the reliability of power reactors in terms of pressure tube/calandria tube contact and deuterium pickup. In addition to accurate knowledge of operating conditions, deformation models are required to represent the development of contact for reliability analysis.

Work Planned for 1990:

1. Develop model for temperature dependence of growth.

2. Modify CDEPTH to do retube calculations.

3. Continue development of shell model for fuel channel including contact algorithm.

4. Complete compilation of diameter, sag and contact data base.

5. Compare curvature and diameter predictions to measured curvatures and diameters.

6. Continue assessment of relationship between offcut microstructure and elongation rate.

7. Extend contact spreading model to lower powered channels, small diameter spacers and CANDU 600 operating conditions.

7.2 High Fluence Irradiation Deformation Behaviour of Zr-2.5%Nb Pressure Tubes

The purpose of this program is to determine the effects of microstructure, temperature and fast neutron fluence on the irradiation deformation behaviour of Zr-2.5% Nb pressure tube material and to extend the data base for both creep and growth to end-of-life fluences.

An irradiation growth program has been underway in the Dido reactor at Harwell since 1978 to study the effects of microstructure and irradiation conditions on the growth of standard and modified pressure tube materials. The fluence of the most highly irradiated specimens surpassed the average fluence for a central channel in Pickering 3 by mid 1987. However, so far no high fluence creep data have been generated for Zr-2.5% Nb pressure tube material.

To anticipate the possibility of long-term transients in the deformation behaviour of pressure tubes and potential problems late in reactor life it is desirable to obtain both creep and growth data to lifetime fluences. This cannot be done cost effectively in Dido, and a review of available facilities by AECL Research Company and OH Research Division showed the French Osiris Reactor at Saclay to be cost effective for obtaining end-of-life fluences. The Osiris program started in 1989.

Work Planned For 1990

1. Continue irradiation of 40 growth specimens at 280 and 310°C in Dido to 10×10^{25} n/m^2, E>1MeV.

2. Obtain preliminary assessment of effect of grain size on growth.

3. Investigate the microstructures of surveillance tube and irradiation growth specimens by analytical and transmission electron microscopy.

4. Extend assessment of differences in growth of surveillance tube specimens and growth specimens to higher fluences.

5. Assess the effect of temperature changes on long-term growth rates.

6. Continue irradiation of 42 growth and 7 biaxial creep specimens in Osiris at 280°C.

7. Make creep and growth strain measurements after a fluence of 2-3x10^{25} n/m^2, E>1MeV. Prepare an insert for the irradiation of creep and growth specimens to end-of-life fluences at 310°C in Osiris.

8. Measure dimensional changes in specimens irradiated in Osiris for end-of-life fracture assessment.

7.3 Evaluation of Power Reactor Fast Fluxes

The objective of this program is to obtain quantitative correlations between the fast fluxes experienced by power reactor components and by small specimens exposed in test reactors.

Although attempts have been made in the past to relate fast fluxes in different reactors on a displacement damage basis, using calculated fast flux spectra, there is little experimental evidence to validate this approach for fracture and deformation of Zirconium alloys. Some test reactor experiments have given data in marked disagreement with current perceptions of the behaviour of pressure tubes in power reactors. Furthermore, the displacement damage approach may not apply to irradiation hardening and loss of fracture toughness. This situation can be resolved experimentally by irradiating specimens in a power reactor environment. The first bundle irradiations started in B4 in 1989.

Work Planned for 1990:

1. Continue irradiation of two "fixed element" bundles containing creep, growth, tensile and DHC specimens to 100 and 200 MWd/KgU in Bundle #8 position in Bruce 4.

2. Train personnel at CRNL for assembly of bundles to carry pre-irradiated specimens using tooling and procedures developed for this purpose.

3. Analyse displacement damage and flux monitors from first bundle, and test specimens from second and third bundles.

7.4 End-of-Life Fracture Properties of TG3 Pressure Tube Material

The objective of this project is to assess the end-of-life fracture, tensile and delayed hydrogen cracking properties of TG3 pressure tubes by irradiation to end-of-life fluences in the Osiris Reactor in France.

The TG3 pressure tubes were developed for improved dimensional stability and the irradiation growth program at Dido verified that this has been achieved. Nine TG3 tubes were installed in Bruce 8, and TG3 Route 1 is the reference material for future CANDU 3 and CANDU 6 reactors. With the potential for degradation of pressure tubes fracture properties at high fluences, AECL-RC has undertaken a program to assess the end-of-life fracture properties of standard pressure tubes using the Osiris Reactor in France. Our program will provide a direct comparison of the end-of-life fracture properties of TG3 Route 1 pressure tubes with those of standard tubes. The irradiation in OSIRIS started in 1989.

Work Planned for 1990:

1. Conclude Phase 1 and 2 of irradiation of TG3 R1 fracture, tensile and DHC specimens in Osiris.

2. Ship specimens from Phase 1 and possibly phase Phase 2 to CRNL for testing.

3. Test phase 1 specimens and report results.

4. Begin Phase 3 of irradiation.

7.5 Advanced Carrier Bundles

The objective of this project is to assess feasibility of carrier bundles for high exposure irradiations of pressure tube materials in power reactors.

The COG Fuel Channels Program currently funds several million dollars in irradiation programs to assess the effects of the reactor environment on pressure tube properties. The costs of these irradiations are escalating rapidly and the security of the programs is difficult to guarantee (eg, many programs have been set back by delays in establishing the Mark 7 fast neutron rod in NRU as a reliable facility and the necessity to change NRU's calandria in the near future would exacerbate this problem).

Carrier bundles offer the possibility of exposing pressure tube specimens to the power reactor environment for the purposes of assessing its effects on properties. However to be useful for many programs high exposures to the reactor environment are required. This can be achieved only with fast turn around of the specimens. In this project the feasibility of the various aspects of achieving this high turnaround are being assessed.

With 80% availability of the power reactors, it should be possible to achieve long-term rates of irradiation exposure better than those experienced Dido or NRU Mk 7.

Work Planned for 1990

1. Complete feasibility study of advanced carrier bundle concept.

2. Depending on recommendation to COG Fuel Channels Technical Committee in 1989/09, proceed with implementation of advanced carrier bundle concept for Bruce reactors. This will include some or all of modifications to fuel handling system, licensing of enriched bundle and development and testing of bundle design and "hot bundle manufacturing techniques.

8.0 ELECTRON OPTICS

The plan is to continue with the microstructural evaluation of Zr-2.5 Nb pressure tube materials with transmission electron microscopy. The understanding of the microstructure of this material provides input to the development of the understanding of the mechanisms which limit the service life of pressure tubes. Continued access to state-of-the-art instruments will be required for this work and the case to provide such a capability within OHRD will be carefully examined.

The development of the SEM 840A capability will continue along the lines of tying this capability into long-term research programs.

Approved by:

J. Brown
Manager
Metallurgical Research Dept

Prepared by:

R.G. Fleck
Supervising Engineer
Materials Integrity Section

REPORTS

88-325-K	Progress on Probabilistic Assessment of Reactor Cores for Pressure Tube Cracking Due to Hydride Blister Formation *D.V. Leemans, T.P. Byrne and M. Leger*
89-2-K CANDEV 89-55	Creep Response of Fuel Channels in CANDU Nuclear Reactors: Computer Code CDEPTH, Version 6. *R.G. Sauvé and N. Badie*
89-35-P COG 89-103	Model for the Spread of Contact Between Pressure Tube and Calandria Tube. *R.A. Holt and N. Badie*
89-82-P CANDEV 89-56	A Probabilistic Model for Crack Initiation in Zr-2.5 Nb Pressure Tube Materials *W.K. Lee, A.H. Chan, S. Sagat*
89-87-P CANDEV 88-187	Effects of Thermal and Loading History on Delayed Hydride Cracking in Zr-2.5 Nb Alloys *G.K. Shek and H. Seahra*
89-118-P CANDEV 89-92	Dependence of Irradiation Growth of Zr-2.5% Nb Pressure Tube Materials on Operating Conditions *R.A. Holt*
89-141-K	Materials Integrity Section: A Review of 1988 and Proposals for 1989 *R.G. Fleck*
89-153-P CANDEV 89-139	Application of the Non-damaging Notch Concept to the DHC Initiation Problem *W.K. Lee*
89-248-P COG 89-191	Strain Rate Sensitivity and Tensile Properties of High Cold-Worked Zr-2.5 wt% Nb Pressure Tube Material *E.T.C. Ho and M.J. McGraw*
89-271-K	Metallurgy of Pressure Tubes *R.G. Fleck*
M89-51-K	Tensile Properties of 22" Elbow and 21" NPS Darlington PHT Piping Materials at 250°C *E.T.C. Ho and M.J. McGraw*
M89-55-K	Effects of Hydrogen Concentration on Delayed Hydride Cracking Under Load Maneuver Conditions *G.K. Shek and H. Seahra*

REPORTS (Continued)

ME89-1-K Metallographic Examination of Front and Back Offcuts of Tube 541 (P4L20)
G.K. Shek

PRESENTATIONS/PUBLICATIONS

"Model for the Spread of Contact Between Pressure Tube and Calandria Tube", presented at the 10th Bi-annual SMIRT Conference, Anaheim, California, August 15, 1989.
R.A. Holt, N. Badie, S. Unterbergs

"Simulation of Fuel Channel Creep Response in CANDU Nuclear Reactors", presented at the 10th Bi-annual SMIRT Conference, August 18, 1989.
R.G. Sauvé, N. Badie, R.A. Holt

"Intergranular Constraints in Polycrystalline Zircaloy-2", presented at the 10th RISO Symposium on Materials Science Roskilde, Denmark, September 5, 1989.
R.A. Holt, T.M. Holden, A.R. Causey, V. Fidleris.

"Effect of Grain Size Variations on Irradiation Growth", presented at the 10th RISO Symposium on Materials Science, September 6, 1989.
R.A. Holt

"Locating a Leaking Crack by Safe Stimulation", presented at the Leak-Before-Break in Water Reactor Piping and Vessels, Toronto, Ontario, October 24-27, 1989.
C.E. Coleman, S. Sagat, G.K. Shek, D.B. Graham, M.A. Durand

"Contribution of Irradiation Growth to Pressure Tube Deformation, presented at the Fuel Channel Technology Seminar, Alliston, Ontario, November 13-16, 1989.
R.A. Holt

"Constitutive Equation Describing the In-Reactor Deformation of CANDU Pressure Tubes and Calandria Tubes", presented at the Fuel Channel Technology Seminar, Alliston, Ontario, November 13-16, 1989.
A.R. Causey, V. Fidleris, R. Holt

"TG3 Pressure Tube Cracking Problem", presented at the Fuel Channel Technology Seminar, Alliston, Ontario, November 13-16, 1989.
R.A. Holt, G.D. Moan, E.T.C. Ho

"Simulation of Fuel Channel Deformation: Methods and Applications", presented at the Fuel Channel Technology Seminar, Alliston, Ontario, November 13-16, 1989.
R.G. Sauvé, N. Badie, R.A. Holt

"The Fatigue Behaviour of Hydrided Zr-2.5 wt% Nb Pressure Tube Material", presented at the Fuel Channel Technology Seminar, Alliston, Ontario, November 13-16, 1989.
D.A. Scarth and W.K. Lee

"A Probabilistic Model for Crack Initiation in Zr-2.5 wt% Nb Pressure Tube Material", presented at the Fuel Channel Technology Seminar, Alliston, Ontario, November 13-16, 1989.
W.K. Lee, A.H. Chan, S. Sagat

"Crack Tunnelling in Zr-2.5 Nb Pressure Tube by Delayed Hydride Cracking", presented at the Fuel Channel Technology Seminar, Alliston, Ontario, November 13-16, 1989.
G.K. Shek, H. Seahra

"Delayed Hydride Cracking and Fracture - Recent Results", presented at the 1989 Fuel Channel Technology Seminar, Alliston, Ontario, November 13-16, 1989.
M. Leger

"Hydride Blisters in Pressure Tubes", presented at the Fuel Channel Technology Seminar, Alliston, Ontario, November 13-16, 1989.
M. Leger

"Contribution of Irradiation Growth to Pressure Tube Deformation", presented at the Soviet/Canadian Seminar on Pressure Tube Technology, USSR, December 12-16, 1989.
R.A. Holt

"An Overview of the Studies in Irradiation Damage and Deformation in Support of the CANDU Reactors presented at the Soviet/Canadian Seminar on Pressure Tube Technology, USSR, December 12-16, 1989.
R.G. Fleck

"The Measurement of the Deuterium Concentration Distributions in Deuteride Blisters on Zirconium Alloy Pressure Tube Material" - published in the Journal of Nuclear Materials, Vol 165; p156-163 (1989)
T. Laursen, M. Leger, Xin-Pei, MA, J.D. MacArthur, G.R. Palmer & J.L. Whitton

"Applications of Neutron Diffraction to Engineering Problems", published in the Materials Science Forum; Vol 27/28 (1988) pp 359-370.
T.M. Holden, J.H. Root, V. Fidleris, R.A. Holt, and G. Roy.

"The Effect of Temperature on the Irradiation Growth of Cold Worked Zr-2.5 wt% Nb", published in Zirconium in the Nuclear Industry: Eighth Symposium ASTM 1023 L.F.P. Van Swan and C.M. Euchen, Eds, ASTM Philadelphia, 1989, pp 705-721.
R.A. Holt, R.G. Fleck

CONFERENCE AND TECHNICAL MEETINGS

Visit SEN de Saclay to discuss details of irradiation experiments in OSIRIS, April 5-6, 1989.
R.A. Holt

Attended "Fundamentals of Fracture" 3rd Conference, Irsee, FRG, June 19-24, 1989.
M. Leger

Attended First Canadian Materials Science Conference, Royal Military College, Kingston, Ontario, June 21-23, 1989.
E.T.C. Ho

Attended EPRI Project RP1890-5 Utility Advisory Group Meeting Combustion Engineering, Windsor, Connecticut, USA, June 28, 1989.
E.T.C. Ho

Visit CRNL to discuss COG programs, Chalk River, Ontario, July 10-11, 1989.
R.A. Holt

Attended 10th International Conference on Structural Mechanics in Reactor Technology (SMIRT), Anaheim, California, August 13-18, 1989.
N. Badie

Visit CEA, Saclay, France to discuss contract, August 31-September 1, 1989.
R.A. Holt

Attended 10th International RISO Conference on Materials Science, Roskilde, Denmark, September 3-8, 1989.
R.A. Holt, V. Perovic

Visit AEA Technology Ltd., Harwell Laboratory to discuss contract, Harwell, UK, September 11, 1989.
R.A. Holt

Visit AEA Technology Ltd., to discuss Pressure Tube Performance, Winfrith, Dorchester, UK, September 12, 1989.
R.A. Holt

Attended 4th International Conference on Hydrogen Effects on Material Behaviour, Moran, Wyoming, USA, September 12-16, 1989.
G.S. Shek

Review of Ion Beam Irradiation Experiments, CRNL, Chalk River, Ontario, September 25, 1989.
R.A. Holt, R.G. Fleck

Attended TMS Fall Meeting, Indianapolis, Indiana, October 1-5, 1989.
W.K. Lee, A. Donner

Visit CEA Saclay to review COG funded Irradiation Program in OSIRIS, AERE Harwell to review COG funded Irradiation program in DIDO, Cambridge University to discuss sabbatical project with V. Perovic, October 23-27, 1989.
R.A. Holt

Attended Leak-Before-Break Conference-89, Toronto, Ontario, October 27, 1989.
G.K. Shek, M. Leger

Attended Workshop on Crack Initiation at Hydrides, CRNL, Chalk River, Ontario, November 1, 1989.
W.K. Lee, G.K. Shek, M. Leger

Visit AECL to discuss COG projects, November 6, 1989.
R.A. Holt

Attended 12th Technical Conference on "Method of Reducing Scrap in the Production of Tubes, Bars and Shapes", Sheraton-Sand Key Report, Clearwater, Florida, November 6-7, 1989.
E.T.C. Ho

Attended 1989 Fuel Channel Technology Seminar, Alliston, Ontario, November 13-16, 1989.
G.K. Shek, W.K. Lee, M. Leger, K. Tashiro, P. Vesely, N. Badie, H. Seahra, R.A. Holt, R.G. Fleck

Participate in Soviet-Canadian Seminar on Pressure Tube Technology, Moscow, USSR, December 12-16, 1989.
R.G. Fleck, R.A. Holt

Participate in Workshop on Hydriding of Zr-2.5 Nb Pressure Tubes, Winnipeg, Manitoba, September 20, 1989.
G.K. Shek